UCID-21218 Rev. 2

Packaging Review Guide for Reviewing Safety Analysis Reports for Packagings

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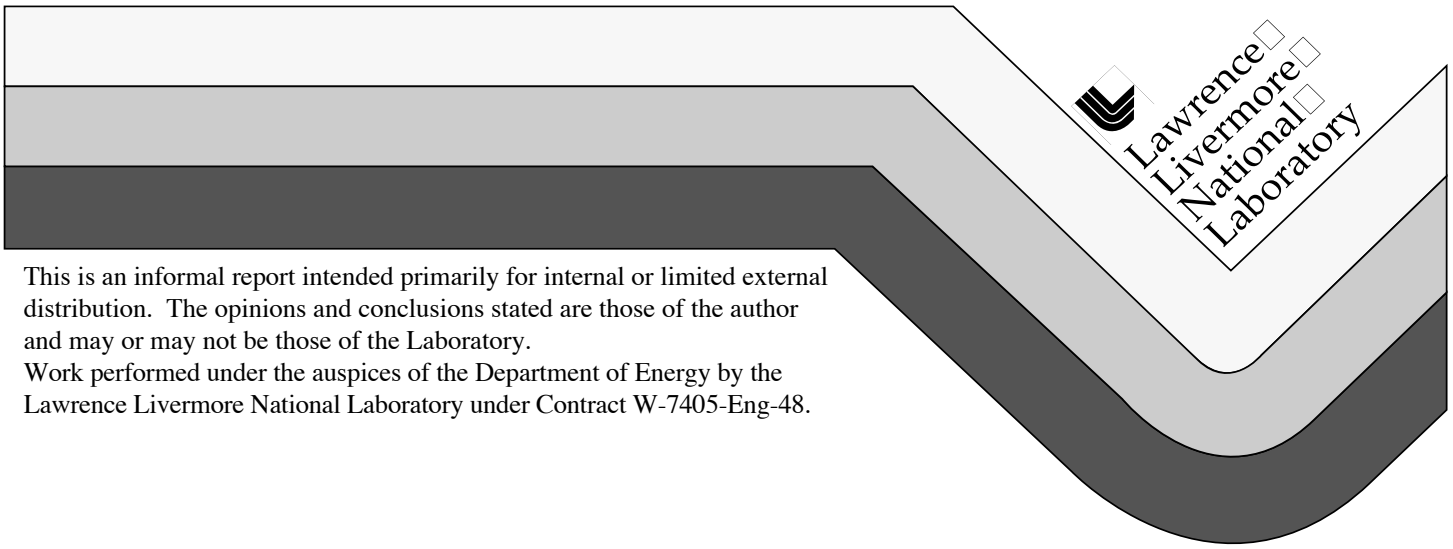

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for Reviewing
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Prepared by

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DISCLAIMER

ABSTRACT

This Packaging Review Guide (PRG) provides guidance for DOE review and approval of packagings to transport fissile and Type B quantities of radioactive material. It fulfills, in part, the requirements of DOE Order 460.1A for the Headquarters Certifying Official to establish standards and to provide guidance for the preparation of Safety Analysis Reports for Packagings (SARPs).

The PRG is intended for use by the Headquarters Certifying Official and his review staff, DOE Secretarial offices, operations/field offices, and applicants for DOE packaging approval.

The PRG is generally organized at the section level in a format similar to that recommended in Regulatory Guide 7.9 (RG 7.9). One notable exception is the addition of Section 9 (Quality Assurance), which is not included as a separate chapter in RG 7.9. Within each section, the PRG addresses the technical and regulatory bases for the review, the manner in which the review is accomplished, and findings that are generally applicable for a package that meets the approval standards.

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ACKNOWLEDGMENT

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Revision 2 is a complete update of the 1988 edition of this document, whose authors deserve acknowledgment for the initial foundation from which this work was developed: L. E. Fischer, C. K. Chou, W. R. Lloyd, M. E. Mount, T. A. Nelson, M. W. Schwartz, and M. C. Witte.

As noted in the Introduction, this document incorporates substantial guidance from various reports published by the U.S. Nuclear Regulatory Commission (NRC), including NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," and NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel." The authors would like to thank the NRC staff, with whom we have worked for more than a decade, for their support and assistance.

Appreciation should also be acknowledged to the review staff at Eagle Research Group, Inc., and Argonne National Laboratory for their comments and suggestions on the draft of this document, especially Bud Fabian for his contributions to the Quality Assurance section.

ABBREVIATIONS AND ACRONYMS

ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel (ASME Code)
Bq	becquerel
cc	cubic centimeter
CFR	U.S. Code of Federal Regulations
cg	center of gravity
Ci	curie
cm	centimeter
CoC	certificate of compliance
DOE	U.S. Department of Energy
DOE O	U.S. Department of Energy Order (used in designation of new-series orders)
DOT	U.S. Department of Transportation
ft.	foot
g	acceleration due to gravity
h	hour
HAC	hypothetical accident conditions
in.	inch
k_{eff}	effective multiplication factor
kPa	kilopascal
LLNL	Lawrence Livermore National Laboratory
m	meter
MPa	megapascal
mrem	millirem

mSv	millisievert
NCT	normal conditions of transport
NRC	U.S. Nuclear Regulatory Commission
PBq	petabecquerel (10^{15} Bq)
PRG	Packaging Review Guide
psi	pounds (force) per square inch
QA	quality assurance
ref	reference
RG	Regulatory Guide
s	second
SARP	Safety Analysis Report for Packaging(s)*
SCO	surface contaminated object
SSCs	structures, systems, and components (important to safety)
SER	Safety Evaluation Report
Sv	sievert
TBq	terabecquerel (10^{12} Bq)
TI	transport index
TRR	Technical Review Report

* The term "SARP" is commonly used by DOE and its contractors to denote the document that describes and evaluates the proposed package. NRC licensees typically use the term "Safety Analysis Report (SAR)." In addition to the SARP, the "application" typically includes a transmittal letter and other supplemental information docketed during the review process.

INTRODUCTION

Background

DOE O 460.1A[1] establishes requirements for the proper packaging and transportation of hazardous material by DOE and its contractors.* Unless otherwise authorized or excluded by this order, DOE transportation of fissile and Type B quantities of radioactive material must be in packagings approved by the Headquarters Certifying Official under conditions specified in the DOE certificate of compliance.

The authority for DOE to certify packagings is established by 49 CFR 173.7(d),[2] which states that packagings made by or under the direction of DOE may be used for the transportation of radioactive materials when evaluated, approved, and certified by DOE against standards equivalent to those specified in 10 CFR Part 71.[3] DOE O 460.1A explicitly states that such packages must comply with the standards of 10 CFR Part 71 and with any other requirements deemed applicable by the Headquarters Certifying Official.

Purpose

This Packaging Review Guide (PRG) provides guidance for DOE review and approval of packagings to transport fissile and Type B quantities of radioactive material. It fulfills, in part, the requirements of DOE O 460.1A for the Headquarters Certifying Official to establish standards and to provide guidance for the preparation of Safety Analysis Reports for Packagings (SARPs).

The PRG is intended for use by the Headquarters Certifying Official and his review staff, DOE Secretarial offices, operations/field offices, and applicants for DOE packaging approval. The primary objectives of the PRG are to:

- Summarize the regulatory requirements for package approval
- Describe the technical review procedures by which DOE determines that these requirements have been satisfied
- Establish and maintain the quality and uniformity of reviews
- Define the base from which to evaluate proposed changes in scope and requirements of reviews
- Provide the above information to DOE organizations, contractors, other government agencies, and interested members of the general public.

The PRG was originally published in September 1987. Revision 1, issued in October 1988, added a review section on quality assurance, special form radioactive material, and penetrations through the containment boundary. Revision 2 of the PRG is a complete update and supersedes Revision 1 in its entirety.

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* Similar requirements were previously established by DOE 1540.2, which may still be applicable depending on specific contractual relationships.

Related Documents

DOE's authority to certify packages is based on the premise that the DOE evaluation and approval process will provide an assurance of safety equivalent to that required by the NRC. Such assurance can be provided by:

- Requiring that DOE package designs meet the standards of 10 CFR Part 71 or their equivalent
- Ensuring that the evaluation methods used to demonstrate compliance with these standards are equivalent to those used by the NRC.

Consequently, the evaluation process described in the PRG relies substantially on 10 CFR Part 71 and the following other NRC documents:

- NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material"[4]
- NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel"[5]
- Regulatory Guide 7.9, "Standard Format and Content of Part 71 Applications for Approval of Packaging for Radioactive Material"[6,7]
- Other regulatory guides and NUREG reports that provide guidance on criteria for evaluating transportation packages.

Scope

Because of the large variety of packages and the many different approaches that can be taken to evaluate these packaging designs, no single guide can address in detail every situation that might be applicable to a review. The PRG is intended to provide a general description of the principles and procedures for evaluating packaging applications. DOE may therefore need to modify or expand the guidance in the PRG to adapt to specific packaging designs. The PRG does not relieve any DOE element or contractor from the requirements of DOE O 460.1A or other pertinent regulations, or imply that SARPs reviewed in accordance with this guide will necessarily be approved.

The PRG addresses shipment of fissile or Type B quantities of radioactive material in DOE certified packagings under the provision of DOE 460.1A and 10 CFR Part 71. The following areas of DOE O 460.1A and 10 CFR Part 71 *are not* currently within the scope of the PRG:

- Shipment of hazardous material other than fissile and Type B radioactive material
- Shipment of DOE radioactive material in packages approved by DOT, NRC, or IAEA
- Shipment of plutonium by air
- Qualification and shipment of low specific activity material and surface contaminated objects
- Qualification and shipment of special form radioactive material

- Notifications, violations, and penalties
- Exemptions and exceptions
- Requirements incorporated into DOE O 460.1A or 10 CFR Part 71 by reference to other regulations (e.g., DOE, NRC, DOT, or U.S. Postal Service).

Organization of PRG

The main body of the PRG is organized into nine sections in a format similar to that recommended in Regulatory Guide 7.9 (RG 7.9) for the SARP.* One notable exception is the addition of Section 9 (Quality Assurance), which is not included as a separate chapter in RG 7.9. Within each section, the PRG addresses the technical and regulatory bases for the review, the manner in which the review is accomplished, and general findings applicable to a package that meets the approval standards. Each section follows the format below.

Introduction

The introduction succinctly states the objective of the review for each section, provides summary information as appropriate, and relates the review to information provided in other chapters of the SARP.

No chapter of a SARP can be reviewed independently from information presented in other chapters. For example, the Containment review depends in part on (1) the packaging and contents description presented in the General Information chapter and (2) the condition of the package under the normal and hypothetical accident condition tests in the Structural and Thermal Evaluation chapters. Likewise, the results of the Containment review may result in the need to implement specific Operating Procedures, Acceptance Tests, or other Quality Assurance procedures. The introduction to each section of the PRG presents a schematic representation of these interfaces. These representations are intended only as examples and should not be considered as a complete list of all information to be reviewed. In addition, specific interfaces may vary with the details of a particular package design or with the specific format of the SARP.

Subsection 1. Areas of Review

This subsection identifies the principal areas that are reviewed to demonstrate that the packaging design complies with regulatory requirements. In general, the Areas of Review correspond to the major subsections of RG 7.9 although in some cases they have been modified for clarity and completeness.

Subsection 2. Regulatory Requirements

This subsection summarizes the applicable regulatory requirements of 10 CFR Part 71. In many instances, the wording from the regulation is shortened, and two or more related requirements are sometimes combined for brevity. This modification in wording is not intended to change or interpret the regulations. Furthermore, the reader is cautioned that the categorization of regulatory requirements by SARP section (or PRG chapter) is a subjective judgment, which may depend on the package design as well as the specific format in which the SARP is organized.

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* For clarification, the major divisions of RG 7.9 (and a SARP) are referred to as "chapters." The major divisions of the PRG are considered "sections."

Regulatory requirements are generally listed in the order that they are addressed in the Review Procedures.

Subsection 3. Review Procedures

This subsection provides guidance for the review of a package. The Review Procedures are organized in parallel with the Areas of Review identified in Subsection 2 above. Because of the large number of different package designs, DOE may need to expand or modify these procedures to adapt to a specific package or to address the method of evaluation presented in the SARP.

The review of the evaluation presented in the SARP will often necessitate a confirmatory analysis by the reviewers. The effort and level of detail of such analysis will depend on many factors, including the issue evaluated, the method of evaluation (e.g., test or analysis), the complexity of the evaluation, the experience of the reviewer, similarity to other approved packages, the margin between evaluated performance and regulatory requirements, importance to safety, and many other factors.

Subsection 4. Evaluation Findings

This subsection presents an example of the major findings of the review. The review staff will modify the wording as appropriate to address specific details of the SARP and methods of review. In addition, this subsection identifies typical limiting assumptions or conditions applicable to the evaluation that might not be specified in the General Information chapter of the SARP but that should be included as conditions of approval in the certificate of compliance.

Subsection 5. References

This subsection identifies references cited in the section. DOE orders are specified in the PRG by order number (e.g., DOE O 460.1A or DOE 5700.6C). Revision designations (e.g., A, B, C) are those in effect at the time of publication of the PRG.

Appendices of PRG

The PRG contains four appendices. Appendix A provides definitions of common package-related terms, many of which are also defined in 10 CFR Part 71 or 49 CFR Part 173. Appendix B presents a summary listing of 10 CFR Part 71 requirements and the SARP chapters to which they are generally applicable. The 1996 revision of 10 CFR Part 71 resulted in several changes and additional requirements, which are highlighted in Appendix C. A summary of issues relevant to materials and fabrication, which are typically addressed in several SARP chapters, is included in Appendix D.

Requirements and Guidance

Throughout the PRG, the word *must* is intended to imply a requirement imposed by CFR or DOE order. Other conditions generally considered necessary for package approval are specified by the word *should*. Because these conditions are not specifically imposed by regulation or order, the SARP may, if appropriate, justify that they are not applicable or that other conditions are more pertinent to the proposed package.

Technical Review Report

The technical review of DOE SARPs is conducted by Lawrence Livermore National Laboratory (LLNL) or Argonne National Laboratory (ANL). The results of the these reviews are documented in a Technical Review Report (TRR) which summarizes:

- Applicable regulatory requirements
- Methods by which the SARP demonstrated that these requirements were met
- A description of the technical review of the evaluation presented in the SARP, including confirmatory analysis and other bases for accepting the SARP evaluation
- Summary findings of the technical review.

The TRR provides the justification for the technical information included in the Safety Evaluation Report (SER), a report issued by the Headquarters Certifying Official to document DOE's review of the package for compliance with DOE O 460.1A and 10 CFR Part 71.

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1. U.S. Department of Energy, "Packaging and Transportation Safety," DOE Order 460.1A, October 2, 1996, as revised.
2. U.S. Code of Federal Regulations, "Shippers—General Requirements for Shipments and Packagings," Part 173, Subchapter A, Chapter I, Title 49, "Transportation."
3. U.S. Code of Federal Regulations, "Packaging and Transportation of Radioactive Material," Part 71, Chapter I, Title 10, "Energy."
4. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Transportation Packages for Radioactive Material," NUREG-1609, May 1999.
5. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," NUREG-1617, Draft Report for Comment, March 1998.
6. U.S. Nuclear Regulatory Commission, "Standard Format and Content of Part 71 Applications for Approval of Packaging for Radioactive Material," Task FC 416-4, Division 7, Proposed Revision 2 to Regulatory Guide 7.9.
7. U.S. Nuclear Regulatory Commission, "Standard Format and Content of Part 71 Applications for Approval of Packaging of Type B, Large Quantity, and Fissile Radioactive Material Packages," Regulatory Guide 7.9, Rev. 1.

1 GENERAL INFORMATION REVIEW

This review verifies that the package design has been described in sufficient detail to provide an adequate basis for its evaluation.

The General Information chapter of the SARP is reviewed by all members of the review team. During the review, the team leader (or his designee) coordinates input from team members and prepares questions or requests for additional information from the applicant as appropriate. At the completion of the review, the individual responsible for questions on the General Information chapter also prepares the corresponding section of the TRR.

The results of the General Information review are considered in the review of all other chapters of the SARP. An example of this information flow for this review is shown in Figure 1-1.

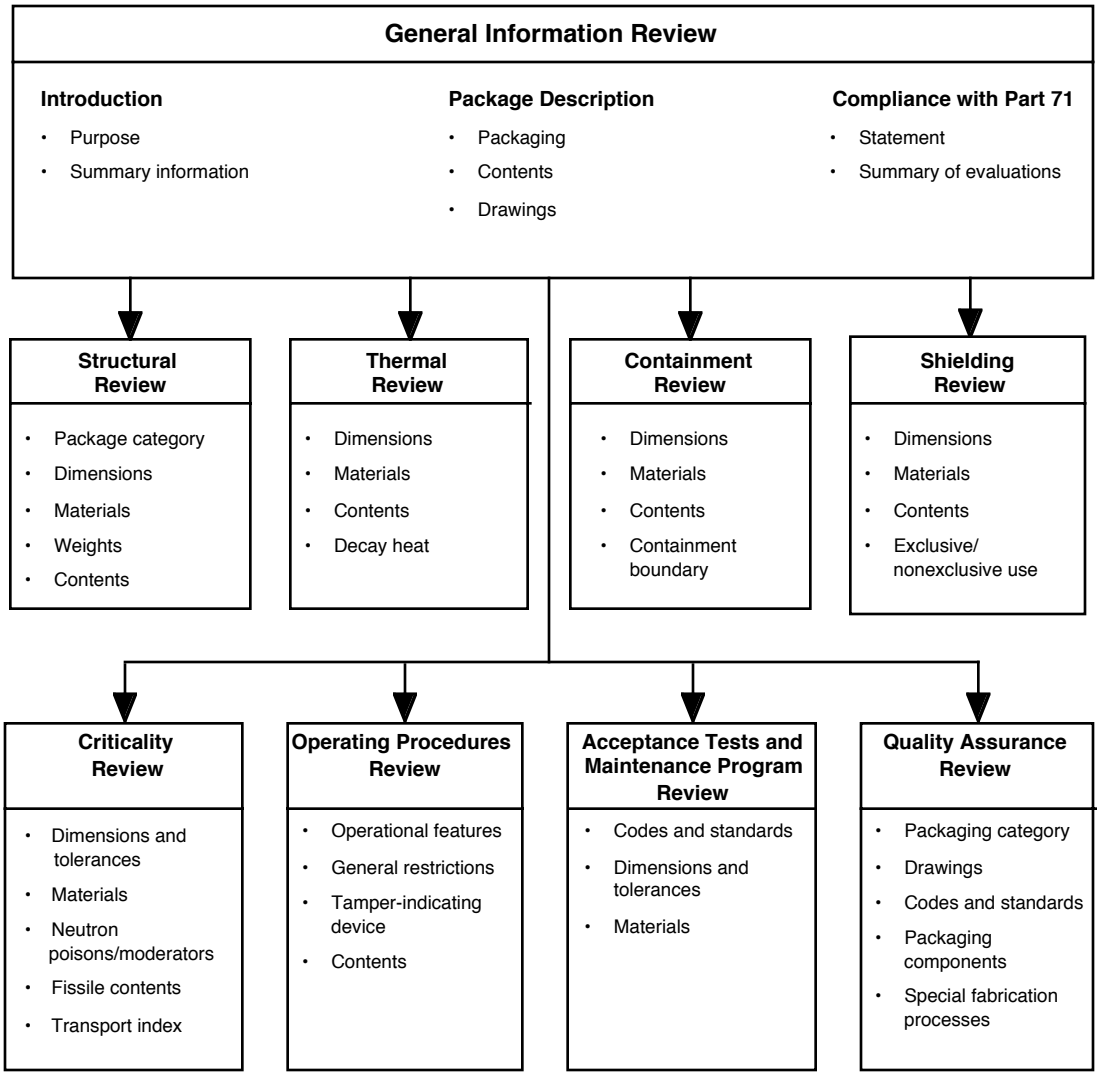

Figure 1-1 Example of Information Flow for the General Information Review

1.1 Areas of Review

The package description and engineering drawings should be reviewed. The review should include:

1.1.1 Introduction

- Purpose of Application
- Summary Information

1.1.2 Package Description

- Packaging
- Contents

1.1.3 Compliance with 10 CFR Part 71

- Statement of Compliance
- Summary of Evaluation

1.1.4 Appendix

- Drawings
- Other Information

1.2 Regulatory Requirements

The requirements of 10 CFR Part 71 applicable to the General Information review include:

- An application for package approval must be submitted in accordance with Subpart D of 10 CFR Part 71. [§71.0(d)]
- An application for modification of a previously approved package is subject to the provisions of §71.13 and §71.31(b). All changes in the conditions of package approval must be approved. [§71.13, §71.31(b), §71.107(c)]
- An application for renewal of a previously approved package must be submitted no later than 30 days prior to the expiration date of the approval to assure continued use. [§71.38]
- The maximum activity of radionuclides in a Type A package must not exceed the limits of 10 CFR Part 71, Table A-1. For a mixture of radionuclides, the provisions of Appendix A, paragraph IV apply, except that for krypton-85, an effective A_2 equal to 10 A_2 may be used. [Appendix A, §71.51(b)]
- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]

- The application must reference or describe the quality assurance program applicable to the package. [§71.31(a)(3), §71.37]
- A fissile material package must be assigned a transport index for nuclear criticality control to limit the number of packages in a single shipment. [§71.59, §71.35(b)]
- A package with a transport index greater than 10 or an accessible external surface temperature greater than 50°C (122°F) must be transported by exclusive-use shipment. [§71.47(a), §71.47(b), §71.59(c), §71.43(g)]
- The application must include a description of the packaging design in sufficient detail to provide an adequate basis for its evaluation. [§71.31(a)(1), §71.33(a)]
- A package for the shipment of plutonium must satisfy the special containment requirements of §71.63(b).
- The smallest overall dimension of the package must not be less than 10 cm (4 in.). [§71.43(a)]
- The outside of the package must incorporate a feature that, while intact, would be evidence that the package has not been opened by unauthorized persons. [§71.43(b)]
- The application must include a description of the contents in sufficient detail to provide an adequate basis for evaluation of the packaging design. [§71.31(a)(1), §71.33(b)]
- Plutonium in excess of 0.74 TBq (20 Ci) must be shipped as a solid. [§71.63(a)]
- The package must be conspicuously and durably marked with its model number, serial number, gross weight, and package identification number. [§71.85(c), §71.13]

1.3 Review Procedures

The following procedures are generally applicable to the review of the General Information chapter of the SARP. These procedures correspond to the Areas of Review listed in Section 1.1 of this PRG.

1.3.1 Introduction

1.3.1.1 Purpose of Application

Verify that the purpose of the application is clearly stated. The application may be for approval of a new design, for modification of an approved design, or for renewal of an existing approval (e.g., certificate of compliance). The purpose may be identified in the SARP itself, or in an accompanying transmittal letter for the application.

Applications for approval of a new design should be complete and should contain the information identified in Subpart D (Application for Package Approval) of 10 CFR Part 71.

Applications for modification of an approved design should clearly identify the changes being requested. Modifications may include design changes, changes in authorized contents, or changes in the conditions of the approval (including changes in the designation of the package identification number). Design changes should be clearly identified on revised engineering

drawings. The application should include an assessment of the requested changes and justification that these changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71. Applications for modifications are subject to the provisions of §71.13 and §71.31(b), as applicable. Changes in the package identification number to designate compliance with revised regulations (e.g., the addition of "-85") are subject to §71.13(d). A summary of regulatory changes affecting the "-85" designation are provided in Appendix C of this PRG.

Applications for renewal of an existing approval should be made within 30 days of expiration of the approval to assure continued use. Expiration of approvals and applications for renewal are subject to the provisions of §71.38.

1.3.1.2 Summary Information

Confirm that the package type and model number are designated. A new Type B package design should be designated B(U)-85 unless it has a maximum normal operating pressure greater than 700 kPa (100 psi) gauge or a pressure relief device that would allow the release of radioactive material under the tests specified in §71.73 (hypothetical accident conditions). In those cases, the package should be designated B(M)-85.

Review the maximum activity and radionuclides of the contents. Ensure they are consistent with the designated package type. For a mixture of radionuclides, the maximum activity allowed in a Type A package must be determined in accordance with 10 CFR Part 71 Appendix A and §71.51(b). Packages for transporting fissile radionuclides should also be designated as fissile material packages (e.g., AF-85, B(U)F-85) unless the exemptions of §71.53 are applicable.

Ensure that any restrictions regarding the type of conveyance for shipment of the package are designated. Note that special requirements apply to the air shipment of plutonium, e.g., §71.64, §71.74, and §71.88. Review of packagings for plutonium air shipments is not addressed in the PRG.

For Type B packages, verify that the designated package category is properly justified. Definitions of package categories are summarized in Table 1.1. Detailed justification, including calculation of an effective A_2 from the maximum activity of the contents, might be presented in the appendix to the General Information chapter or in another chapter of the SARP (e.g., Containment).

Table 1.1 Category Designations for Type B Packages[1]

Contents Form	Category I	Category II	Category III
Normal Form*	Greater than 3,000 A_2 or greater than 1.11 PBq (30,000 Ci)	Between 3,000 A_2 and 30 A_2, and not greater than 1.11 PBq (30,000 Ci)	Less than 30 A_2 and less than 1.11 PBq (30,000 Ci)

* Similar requirements apply to special form radioactive material, which is not explicitly addressed in this PRG.

The package category will determine which code[2] or other criteria[3,4] are appropriate for components that affect the structural integrity of containment, criticality, or shielding systems. Although the designation of these codes or standards should be indicated on the engineering drawings and applicable fabrication specifications in this chapter (see Section 1.3.4.1), a more detailed discussion and justification may be deferred to the Structural Evaluation chapter of the SARP. Similarly, details of other codes and standards for the package may be presented in the General Information chapter or may be discussed in the applicable chapter of the SARP. Review designated codes and standards as appropriate.

Confirm that the SARP identifies the applicant's quality assurance (QA) program applicable to the package. Details of QA program requirements should be presented in the QA chapter of the SARP.

For fissile material packages, confirm that a transport index (TI) based on nuclear criticality safety is designated for each contents. This index will generally be designated in the certificate of compliance as the *minimum transport index*. Note that the TI used in shipment depends on both criticality safety and external radiation levels. Unlike the TI based on criticality, the TI based on radiation is determined by radiation levels of the package as loaded for shipment and is not specified in the certificate of compliance. Ensure that the maximum number of packages that may be shipped in a single conveyance and any restrictions for exclusive-use shipment, if applicable, are consistent with the TI based on criticality safety.

Determine if the shipment of the package is limited to exclusive use because of other regulatory requirements (e.g., external radiation levels or package surface temperatures). Additional information should be included in the Operating Procedures chapter of the SARP.

1.3.2 Package Description

1.3.2.1 Packaging

Review the text description of the packaging. Sketches, figures, or other schematic diagrams should be provided as appropriate. Ensure that the description of the packaging presented in text and figures is consistent with that depicted on the engineering drawings (see Section 1.3.4.1).

Verify that the following information, as applicable, is adequately discussed:

- General packaging description, including overall dimensions, maximum weight, and minimum weight, if appropriate
- Containment features, including a clear identification of the containment boundary
- Shielding features, including personnel barriers
- Criticality control features, including neutron poisons, moderators, and spacers
- Heat-transfer features, including gaps and coolants, that affect transfer and dissipation of heat
- Structural features, including supporting structures, lifting and tie-down devices, and impact limiters.

Proprietary information, if applicable, should be clearly identified. Justification for withholding this information from public disclosure should be presented in a format comparable to that specified in 10 CFR 2.790.

Verify that the SARP defines the exact boundary of the containment system. This may include the containment vessel, welds, drain or fill ports, valves, pressure relief devices, seals, test ports, lids, cover plates, and other closure devices. If multiple seals are used for a single closure, the seal defined as the containment-system seal should be clearly identified. A sketch of the containment system should be provided, and all components should be shown on the engineering drawings in the appendix. If the contents include plutonium in excess of 0.74 TBq (20 Ci), the packaging must have both an inner and outer containment system unless exempted by §71.63(b). Additional information regarding the review of the containment boundary and special containment requirements for damaged reactor fuel are addressed in Section 4 of the PRG.

Based on the package description and engineering drawings, confirm that the package meets the following requirements of §71.43:

- The smallest overall dimension of the package is not less than 10 cm (4 in.)
- The outside of a package incorporates a feature that, while intact, is evidence that the package has not been opened by unauthorized persons.

Verify that appropriate operational features are discussed. A schematic diagram of any special operational feature should be included if applicable. Additional information on operational features may be presented in the Operating Procedures chapter of the SARP.

1.3.2.2 Contents

Confirm that the contents are described in the same detail as that intended for the certificate of compliance. The description should include, as a minimum, the following information:

- Identification and maximum quantity (radioactivity or mass) of the radioactive material
- Identification and maximum quantity of fissile material
- Chemical and physical form, including density and moisture content, and the presence of other moderating constituents
- Location and configuration of contents within the packaging, including secondary containers, wrapping, shoring, and other material not defined as part of the packaging
- Identification and quantity of nonfissile materials used as reflectors, neutron absorbers, or moderators
- Any material subject to chemical, galvanic, or other reaction, including the generation of combustible and reactive gases
- Maximum normal operating pressure
- Maximum weight (including secondary containers, shoring, etc.) and minimum weight if appropriate
- Maximum decay heat.

If the contents include reactor fuel rods or assemblies, the following additional information should be specified as appropriate:

- Type of fuel, maximum enrichment and density of fissile material prior to irradiation (including specifications of non-uniform enrichment, if applicable). If the reactivity of irradiated fuel is larger than fresh fuel, the isotopic composition of the irradiated fuel should also be presented.
- Burnup, minimum initial enrichment, specific power, cooling time, and heat load
- Fuel assembly specifications, including dimensional data for the fuel pellets, cladding, fuel-cladding gap, rods, guide tubes, and other assembly structures considered in the evaluation
- Control assemblies or other contents (e.g., startup sources) present
- Number of assemblies or rods
- For damaged fuel, the extent of damage, description of containerization, or any other applicable limits
- Other information necessary to evaluate compliance with 10 CFR Part 71, as applicable.

If the contents include plutonium in excess of 0.74 TBq (20 Ci), verify that the contents are in solid form.

1.3.3 Compliance with 10 CFR Part 71

1.3.3.1 Statement of Compliance

Confirm that SARP contains an unequivocal statement that the package complies with 10 CFR Part 71.

1.3.3.2 Summary of Evaluation

In addition to a statement that the package complies with 10 CFR Part 71, the General Information chapter of the SARP should include a summary of the package evaluations presented in subsequent SARP chapters, with a specific reference to the chapters in which compliance is demonstrated. The summary information should address:

- Structural and thermal performance of the package under the tests for normal conditions of transport and hypothetical accident conditions, §71.71 and §71.73, respectively
- General requirements for all packages, §71.43
- Structural requirements for lifting and tie-down devices and for irradiated nuclear fuel shipments, §71.45 and §71.61
- External radiation requirements for all packages, §71.47
- Requirements for Type B packages, §71.51
- Criticality requirements, §71.53, §71.55, §71.59

- Special requirements for plutonium packages, §71.63
- Requirements for operating controls and procedures, Subpart G
- Requirements for quality assurance, Subpart H.

The review of each SARP chapter should confirm that this summary information is consistent with the detailed evaluation and with the requirements of 10 CFR Part 71.

1.3.4 Appendix

1.3.4.1 Drawings

Verify that information on the engineering drawings is sufficiently detailed and consistent with the package description. A detailed discussion of information to be included on drawings is presented in NUREG/CR-5502.5

DOE orders (e.g., DOE O 460.1A and 1540.2) authorize transportation of Type B or fissile radioactive material by DOE and DOE contractors in packages approved by the Headquarters Certifying Official under conditions specified in the certificate of compliance. The purpose of the engineering drawings in the SARP is to define the package design approved by DOE, and compliance with these drawings is typically included in the certificate as a condition of package approval. Packages that do not conform to the drawings in the SARP are not authorized for use.

Confirm that each drawing has a title block that identifies the preparing organization, drawing number, sheet number, title, date, and signature or initials indicating approval of the drawing. Revised drawings should identify the revision number, date, and description of the change in each revision. Proprietary information, if applicable, should be clearly identified. The drawings should include:

- General arrangement of packaging and contents, including dimensions
- Design features that affect the package evaluation (see Section 1.3.2.1 above)
- Packaging markings, including model number, serial number, gross weight, and package identification number
- Maximum allowable weight of the package
- Maximum allowable weight of the contents and secondary packaging
- Minimum weights, if appropriate.

Information on design features should include, as appropriate:

- Identification of the design feature and its components
- Materials of construction, including applicable material specifications
- Codes, standards, or other similar specification documents for fabrication, assembly, and testing (including welding symbols). As appropriate, such information may be included on a separate fabrication specification that can be referenced as a condition of approval in

the certificate. Compliance with this specification should generally be noted on the drawings as applicable.

- Location, with respect to other package features
- Dimensions with appropriate tolerances
- Operational specifications (e.g., bolt torque, specifications of pressure-relief devices).

1.3.4.2 Other Information

Confirm that the appendix includes a list of references and a copy of any applicable reference not generally available to the reviewer, as appropriate. The appendix may also provide supporting information on special fabrication procedures (as noted on the drawings), determination of the package category, and other appropriate supplemental information deemed necessary by the applicant or reviewer.

1.4 Evaluation Findings

1.4.1 Findings

The review should ensure that the information presented supports a conclusion that the regulatory requirements in Section 1.2 above are satisfied. Because confirmation of some information presented in the General Information chapter of the SARP depends on a detailed review of subsequent chapters, preparation of the findings for this section may be deferred until the review of later chapters is completed.

The TRR should include a finding similar to the following:

> Based on review of the statements and representations in the SARP, the staff concludes that the package design has been adequately described to meet the requirements of 10 CFR Part 71. This description also demonstrates that the packaging meets the minimum size limitations and contains a tamper-indicating device required by the regulation.

1.4.2 Conditions of Approval

The TRR should clearly identify any conditions of approval that should be included in Section 5 of the certificate of compliance. In addition to a summary package description and specifications of authorized contents, the conditions of approval applicable to the General Information chapter of the SARP typically include:

- Type of conveyance
- Minimum transport index (based on criticality safety)
- Restriction to exclusive-use shipment, if applicable
- Drawings that define the package design, and additional fabrication specifications as applicable
- Requirement to add serial numbers to previously approved packages, as applicable.

1.5 References

1. U.S. Nuclear Regulatory Commission, “Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m),” Regulatory Guide 7.11.
2. American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code*, 1998, New York.
3. U.S. Nuclear Regulatory Commission, “Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials,” NUREG/CR-3019 (UCRL-53044), March 1984.
4. U.S. Nuclear Regulatory Commission, “Fabrication Criteria for Shipping Containers,” NUREG/CR-3854 (UCRL-53544), March 1985.
5. U.S. Nuclear Regulatory Commission, “Engineering Drawings for 10 CFR Part 71 Package Approvals,” NUREG/CR-5502 (UCRL-ID-130438), May 1998.

2 STRUCTURAL REVIEW

This review verifies that the structural performance of the package design has been adequately evaluated for the tests specified under normal conditions of transport and hypothetical accident conditions and that the package design meets the structural requirements of 10 CFR Part 71.

The Structural review is based in part on the descriptions and evaluations presented in the General Information and the Thermal Evaluation chapters of the SARP. Similarly, results of the Structural review are considered in the review of subsequent chapters of the SARP. An example of this information flow for the Structural review is shown in Figure 2-1.

Although 10 CFR Part 71 specifies only a few explicit structural requirements for packages (e.g., lifting and tie-down requirements), the structural performance of the package under normal conditions of transport and hypothetical accident conditions significantly affects its ability to meet the containment, shielding, and subcriticality requirements of the regulation. Consequently, the Structural review focuses on confirming the SARP evaluation of the effects of these tests and on coordinating these effects with the review of the Thermal, Containment, Shielding, and Criticality Evaluation chapters.

2.1 Areas of Review

The structural design of the package should be reviewed. The Structural review should include the following:

2.1.1 Description of Structural Design

- Design Features
- Codes and Standards

2.1.2 Materials of Construction

- Material Specifications and Properties
- Prevention of Chemical, Galvanic, or Other Reactions
- Effects of Radiation on Materials

2.1.3 Fabrication, Assembly, and Examination

- Fabrication and Assembly
- Examination

2.1.4 General Considerations for Structural Evaluations

- Evaluation by Test
- Evaluation by Analysis

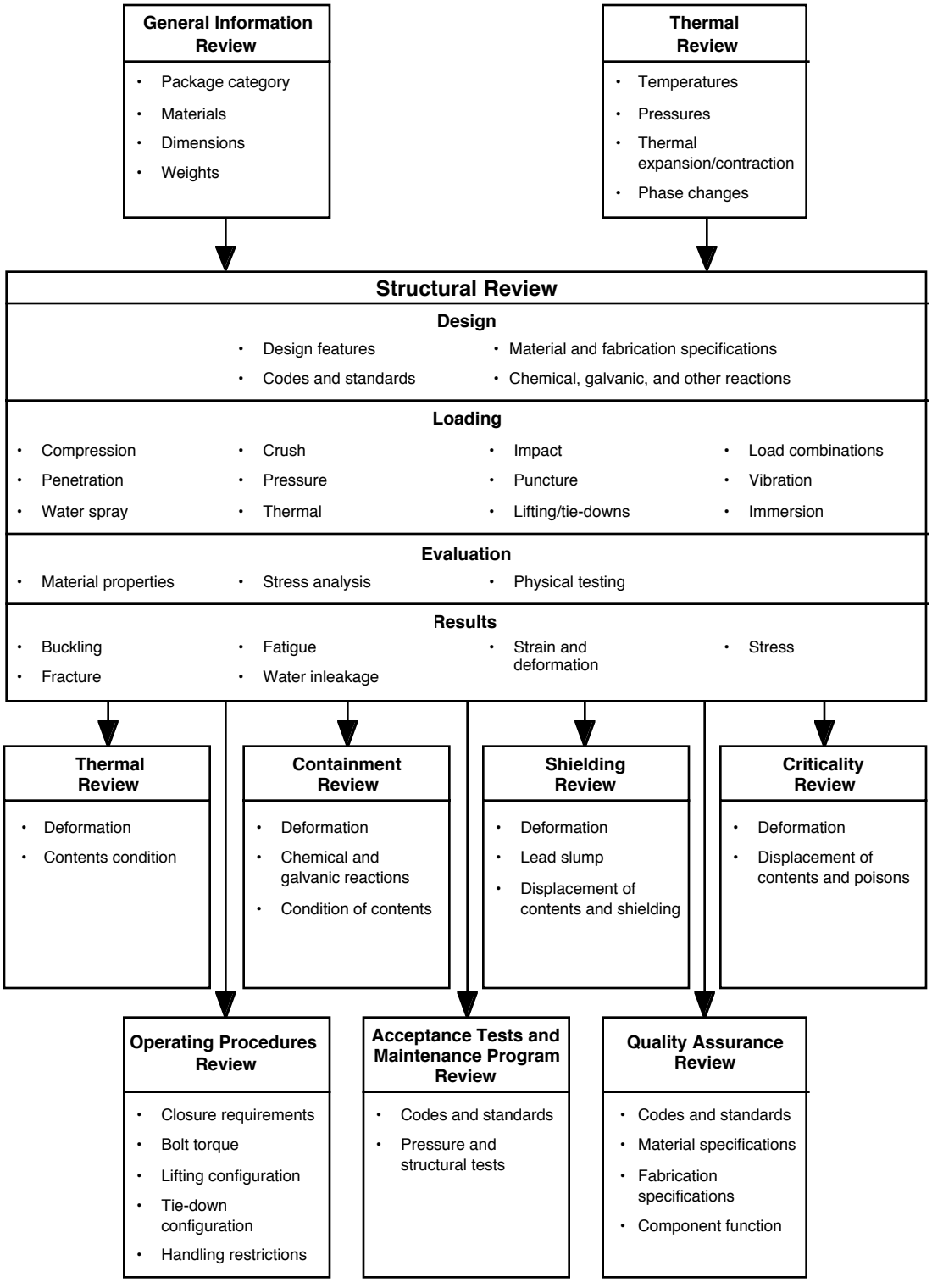

Figure 2-1 Example of Information Flow for the Structural Review

2.1.5 Structural Evaluation for Normal Conditions of Transport

- Heat
- Cold
- Reduced External Pressure
- Increased External Pressure
- Vibration
- Water Spray
- Free Drop
- Corner Drop
- Compression
- Penetration
- Structural Requirements for Fissile Material Packages

2.1.6 Structural Evaluation for Hypothetical Accident Conditions

- Free Drop
- Crush
- Puncture
- Thermal
- Immersion–Fissile material
- Immersion–All packages

2.1.7 Structural Evaluation of Lifting and Tie-Down Devices

- Lifting Devices
- Tie-Down Devices

2.1.8 Structural Evaluation for Special Pressure Conditions

- Special Requirement for Irradiated Nuclear Fuel
- Analysis of Pressure Test

2.1.9 Appendix

2.2 Regulatory Requirements

Regulatory requirements of 10 CFR Part 71 applicable to the Structural review are as follows:

- The package must be described and evaluated to demonstrate that it meets the structural requirements of 10 CFR Part 71. [§71.31(a)(1), §71.31(a)(2), §71.33, §71.35(a)]
- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]
- The package must be made of materials of construction that assure there will be no significant chemical, galvanic, or other reactions, including reactions due to possible inleakage of water, among the packaging components, among package contents, or between the packaging components and the package. The effects of radiation on the materials of construction must be considered. [§71.43(d)]
- The performance of the package must be evaluated under the tests specified in §71.71 for normal conditions of transport. [§71.41(a)]
- The package must be designed, constructed, and prepared for shipment so there would be no loss or dispersal of contents, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging under the tests specified in §71.71 for normal conditions of transport. [§71.43(f), §71.51(a)(1)]
- A package for fissile material must be so designed and constructed and its contents so limited to meet the structural requirements of §71.55(d)(2) through §71.55(d)(4) under the tests specified in §71.71 for normal conditions of transport.
- The performance of the package must be evaluated under the tests specified in §71.73 for hypothetical accident conditions. [§71.41(a)]
- The package design must meet the lifting and tie-down requirements of §71.45.
- A package for irradiated nuclear fuel with a specific activity greater than 37 PBq (10^6 Ci) must be designed so that its undamaged containment system can withstand an external water pressure of 2 MPa (290 psi) for a period of not less than one hour without collapse, buckling, or inleakage of water. [§71.61]
- The package design must have adequate structural integrity to meet the internal pressure test requirement specified in §71.85(b).

2.3 Review Procedures

The following procedures are generally applicable to the review of the Structural Evaluation chapter of the SARP. These procedures correspond to the Areas of Review listed in Section 2.1 of this PRG.

2.3.1 Description of Structural Design

2.3.1.1 Design Features

Review the structural design features presented in the General Information and Structural Evaluation chapters of the SARP. Design features important to the structural evaluation include:

- Components that provide structural integrity for heat transfer, containment, shielding, and subcriticality design features (e.g., impact limiters, containment vessels, neutron-absorber plates)
- Components that affect, or are affected by, the performance of structural components (e.g., lead shielding, lifting and tie-down devices)
- Components that provide structural integrity to the contents (e.g., internal supporting structures).

Information on structural design features should include, as appropriate:

- Location, dimensions, and tolerances
- Materials of construction and their specifications (See Section 2.3.2.1)
- Fabrication methods (See Section 2.3.2.2)
- Weights and centers of gravity of packaging and major subassemblies
- Maximum weight of contents (minimum weight, if appropriate)
- Maximum normal operating pressure
- Description of closure systems
- Description of handling requirements.

Verify that the text and sketches describing the structural design features are consistent with the engineering drawings.

2.3.1.2 Codes and Standards

Confirm that the SARP identifies established codes and standards applicable to the structural evaluation. The codes and standards should be appropriate for the intended purpose and be properly applied. The reviewer should verify that the code or standard:

- Was developed for structures of similar design and material, if not specifically for shipping packages
- Was developed for structures with similar loading conditions
- Was developed for structures that have similar consequences of failure
- Adequately addresses potential failure modes
- Adequately addresses margins of safety.

Several regulatory guides, NUREGs, codes, and standards documents provide guidance for package design. RG 7.8[1] identifies the load combinations to be used in package evaluations, and RG 7.6[2] provides design criteria for containment systems. The criteria of RG 7.6 are based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code,[3] Section III, Division 1, Subsection NB. In addition, ASME has recently published a new code section (Section III, Division 3), which is specifically intended for transportation packages. Although both RG 7.6 and ASME Section III, Division 3, specifically address the containment systems of spent-fuel (and high-level-waste packages), their guidance may also be applied to the

containment systems of other Category I packages. NUREG/CR-4554, Vol. 6[4] and NUREG/CR-6322[5] discuss the buckling evaluation of containment vessels and baskets, respectively. In addition, ANSI N14.6[6] and NUREG-0612[7] have been used for the design of packaging trunnions.

Other NUREGs provide guidance on fabricating package components. NUREG/CR-3854[8] provides a list of industrial codes and standards for fabrication, and NUREG/CR-3019[9] presents criteria specifically for welding. These NUREGs also provide useful guidance for package design because the code or standard for fabrication should be the same as that for design, operation, and maintenance unless justified otherwise.

Table 2.1 summarizes those sections of the ASME B&PV Code that are generally acceptable for Type B packagings, based on the package category designations described in Table 1.1. Because the ASME Code (except for Section III, Division 3) was not developed for transportation packages, various articles may not be applicable and some Code requirements (e.g., pressure relief devices) may not be consistent with 10 CFR Part 71 requirements. The review should ensure that the SARP clearly identifies the provisions of the Code applicable to materials, fabrication, examination, and testing of the packaging and that excluded provisions are appropriately justified. Specifications of Section III, Subsection NB, should be generally be reviewed against those in Section III, Division 3, Subsections WA and WB.

Table 2.1 Sections of ASME B&PV Code Applicable to Type B Packages

Component Function	Category I	Category II	Category III
Containment	Section III, Division 1, Subsection NB or Section III, Division 3	Section III, Division 1, Subsection ND*	Section III, Division 1, Subsection NF**
Criticality (structural support)	Section III, Division 1, Subsection NG (NF for Buckling)		
Shielding and Other Safety Features	Section VIII, Division 1 or Section III, Division 1, Subsection NF		

* Category I criteria are also acceptable.

** Category I and II criteria are also acceptable.

2.3.2 Materials of Construction

Summary guidance for review of materials is presented in Appendix D of this PRG.

2.3.2.1 Material Specifications and Properties

As discussed in Section 1.3.4.1, an appropriate specification should be identified on the engineering drawings for the control of each material. Materials and their properties should be consistent with the design code or standard selected. In the ASME B&PV Code, material specifications are generally addressed in Section II.

Review the properties of the materials of construction. If no code or standard is available, the SARP should provide adequately documented material properties and, as appropriate, justify the quality assurance methods used to ensure that these properties are achieved. Coordinate with the Quality Assurance review as appropriate.

Verify that the material properties are appropriate for the load conditions (e.g., static or dynamic impact loading, hot or cold temperatures, and wet or dry conditions). Confirm that appropriate temperatures at which allowable stress limits are defined are consistent with minimum and maximum service temperatures. Verify that the force-deformation properties for impact limiters are based on appropriate test conditions (e.g., strain rate and temperature). Ensure that materials are thermally stable for long-term exposure at elevated temperatures, as appropriate.

Verify that the materials of structural components have sufficient fracture toughness to preclude brittle fracture under normal conditions of transport and hypothetical accident conditions. RG 7.11[10] and RG 7.12[11] provide criteria for fracture toughness of ferritic steels. Brittle fracture is usually not a concern for austenitic steels unless fabrication processes increase their susceptibility to embrittlement. If the contents include or produce hydrogen gas, ensure that hydrogen embrittlement has been appropriately addressed.

2.3.2.2 Prevention of Chemical, Galvanic, or Other Reactions

Review the materials and coatings of the package to verify that they will not produce a significant chemical, galvanic, or other reaction among packaging components, among packaging contents, or between the packaging components and the package contents. The review should consider reactions resulting from inleakage of water, including wet loading of spent fuel or other contents. Evaluate the possible generation of hydrogen and other flammable or corrosive gases. NRC Information Notice 96-34[12] discusses hydrogen generation that resulted from the reaction between acidic borated water and a zinc coating applied to the internal surfaces of a spent fuel storage cask.

Galvanic interactions and the formation of eutectics should be considered for metallic components that may come into physical contact with one another. Such interactions could occur with depleted uranium, lead, or aluminum in contact with steel.

2.3.2.3 Effects of Radiation on Materials

Verify that the effects of radiation on the packaging materials have been appropriately considered. These effects include degradation of seals, sealing materials, coatings, adhesives, and structural materials.

Review of radiolysis and the associated production of hydrogen and other gases by radiation is discussed in Sections 3 and 4 of the PRG.

2.3.3 Fabrication, Assembly, and Examination

Summary guidance for review of fabrication, assembly, and examination is presented in Appendix D of this PRG.

2.3.3.1 Fabrication and Assembly

10 CFR Part 71 (§71.31(c) and §71.37(a)) specifies that the application should provide information on codes, standards, and the quality assurance program for fabrication and assembly. In terms of the B&PV Code, these processes are referred to as fabrication and installation, and are generally addressed in the 4000-series articles of Section III, with welding qualifications specified in Section IX. In SARP reviews, the term "fabrication" is often used to mean both fabrication and assembly (e.g., welding). As noted above, guidance on appropriate codes and standards is provided in NUREG/CR-3854 and NUREG/CR-3019.

If fabrication and assembly specifications are prescribed by an appropriate code or standard (e.g., ASME, AWS), the code or standard should be identified on the engineering drawings. Unless the SARP justifies otherwise, specifications of the same code or standard used for design should also be used for fabrication and assembly. For components for which no code or standard is applicable, the SARP should identify the specifications on which the evaluation depends and describe the method of control to assure that these specifications are achieved. This description may reference a quality assurance or other appropriate specifications document. Such specifications should be included on the engineering drawings and separate fabrication specifications as appropriate. As noted in Section 1.3.4.1 of this PRG, the engineering drawings are generally specified as conditions of approval in the certificate of compliance.

2.3.3.2 Examination

Although the term "examination" is not specifically mentioned in 10 CFR Part 71, it is generally considered as part of the fabrication and assembly processes, or simply as part of fabrication. In the B&PV Code, examination is addressed in the 5000-series articles of Section III, with additional details on nondestructive-evaluation methods specified in Section V.

Examination addresses the methods and criteria by which the fabrication is determined to be acceptable. Unless the SARP justifies otherwise, specifications of the same code or standard used for fabrication should also be used for examination. For components for which no fabrication code or standard is applicable, the SARP should summarize the examination methods and acceptance criteria in the Acceptance Tests and Maintenance Program chapter. As noted in Section 8 of this PRG, acceptance tests are generally included as conditions of approval in the certificate of compliance. Examination specifications should also be provided on the engineering drawings and fabrication specifications as appropriate.

2.3.4 General Considerations for Structural Evaluations

Structural evaluations of the package design may be performed by analysis, test, or a combination of both methods. The evaluations should demonstrate that the structural performance of the package meets the criteria discussed in Section 2.3.5 below for normal conditions of transport and in Section 2.3.6 for hypothetical accident conditions. Additional conditions for evaluation of the structural design are described in Sections 2.3.7 and 2.3.8. The review of these evaluations should verify that:

- The most unfavorable initial loading and environmental conditions have been addressed. See RG 7.8 for guidance on selection of initial conditions.

- The most unfavorable drop or loading orientations for the entire sequence of tests have been considered. The most unfavorable orientations for one component may not be the most unfavorable for another component.
- The evaluation methods are appropriate for the loading conditions considered and follow accepted practices and precepts.
- The results are interpreted correctly.

2.3.4.1 Evaluation by Test

If the package is evaluated by test, the review should include the following:

- Verify that the test procedures and equipment are adequate. Confirm that the methods and instruments are sufficient for describing the structural response or damage. Both interior and exterior damage should be considered. UCRL-ID-121673[13] provides guidance for drop testing, including the use of reduced-scale models.
- Review the description of the target surface (e.g., material, mass, dimensions) used for the drop, crush, and puncture tests. Confirm that it represents an essentially unyielding surface. An example of such a surface is described in IAEA Safety Series No. 37,[14] but the determination that a surface is essentially unyielding depends on package-specific details.
- Review the description of the steel plate (e.g., material, mass, dimensions, orientation) used for the crush test, if applicable. Confirm that it meets the specifications of §71.73(c)(2).
- Review the description of the steel bar (e.g., material, dimensions, orientation, method of mounting) used for the puncture test. Confirm that it is securely attached to an essentially unyielding surface, has sufficient length to cause maximum damage to the package, and meets the other specifications of §71.73(c)(3).
- Verify that the test specimen has been fabricated using the same materials, methods, and quality assurance as specified in the package design. Any differences should be identified and the effects evaluated in the SARP. The test specimen should include all components that are expected to have significant effects on the test results. Substitutes for the contents and other simulated components should have the same weight, structural properties, and interaction with the packaging as the actual contents and components. If applicable, verify that the scale-model specimen is properly scaled, fabricated, and instrumented. Confirm that the SARP justifies that size effects are not significant (e.g., material properties are not affected by size).
- Verify that the tests consider the orientations for which the most unfavorable damage is expected, and that the selection is justified. The SARP should address drops that (1) produce the highest g-loads on package components and (2) challenge the most vulnerable orientations and components of the package (e.g., bolts, closure rings, seals, valves, and ports). The first group of drops includes those with the package center of gravity (cg) located directly above the center of the impact area, such as end drops, side drops, and cg-over-corner drops. It also includes slap-downs, in which the cg is not

directly over the impact area, as slap-down drops of a long package can produce a high g-load in the second impact. Drops in the second group will depend on the vulnerable package components and their failure modes. Components vulnerable to impact loads should generally be protected by special design features such as recessed construction, protective cover plates, and impact limiters. Ensure that the evaluation of most unfavorable damage considers the thermal (fire) test and water immersion test (if applicable), which follow the drop, crush (if applicable), and puncture tests.

- Verify that the test addresses movement or damage of the contents as appropriate. For example, movement or damage of fuel rods or assemblies may impact the criticality evaluation.
- Verify that all test results are evaluated and their implications interpreted, including interior and exterior damage of the test article. Unexpected or unexplainable test results indicating possible testing problems or non-reproducible specimen behavior should be discussed and evaluated.
- Verify that the interpretation of the test results addresses differences between test conditions and regulatory conditions. For example, ambient temperature and decay heat may result in package temperatures and stresses during transportation that differ from those of the tested specimen.
- Review the video and photos of the tests as appropriate.
- Verify that the test results are reliable and repeatable. Test results should convincingly show that any package fabricated in accordance with the approved design will meet regulatory requirements.
- Review the criteria for evaluating pass/fail for the test conditions. Compare the test results with these criteria. If acceptance tests are performed after the structural testing, the acceptance tests should be performed according to appropriate codes and standards.

2.3.4.2 Evaluation by Analysis

If the package is evaluated by analysis, the review should include the following:

- Verify that the SARP clearly describes the analysis methods, models, and results, including all assumptions and input data. (See RG 7.6 for guidance on design criteria for analysis.)
- Verify that the models and material properties are appropriate for the load combinations considered. Ensure that the material properties (e.g., elastic, plastic) are consistent with the analysis methods. The SARP should justify the strain rate at which the properties were determined. Confirm that the analysis considers true stress-strain or engineering stress-strain, as applicable.
- Verify that the applied boundary conditions in the analysis model are appropriate. For free-drop impact analyses, impact loads for package components are usually derived from the dynamic analyses of the package and used in a quasi-static stress analysis of the component. Confirm that a dynamic amplification factor has been appropriately applied

to account for vibration and other dynamic effects. A summary of quasi-static and dynamic analysis methods for impact analysis is provided in NUREG/CR-3966.[15]

- Verify that the analysis evaluates the most unfavorable orientations, and that the selection is justified. Ensure that the evaluation of most unfavorable damage considers the entire sequence of tests.
- Verify that the analysis evaluates the effect of the test conditions on the contents as appropriate. (See Section 2.3.4.1.)
- Verify that the computer codes, if applicable, are properly used, benchmarked, and maintained under an appropriate quality assurance program. At least one representative input and output file (or key section of the file) should generally be included in the SARP.
- Verify that the response of the package to loads, in terms of stress and strain to components and structural members, is shown and that the structural stability of individual members, as applicable, is evaluated.
- Verify that the results are correctly interpreted and demonstrate adequate margin of safety. The maximum stresses or strains should be compared to corresponding design-code allowables.

2.3.5 Structural Evaluation for Normal Conditions of Transport

The evaluation of the package under the normal conditions of transport is based on the effects of the tests and conditions specified in §71.71. These tests must not result in a significant decrease in package effectiveness. For example, these tests should result in:

- No significant decrease in the effectiveness of packaging components that provide heat transfer or insulation. Coordinate with the Thermal review.
- No significant decrease in the effectiveness of packaging components that provide containment, including no loss or dispersal of contents or release of radioactive material exceeding the requirements of §71.51(a)(1), as applicable. Coordinate with the Containment review.
- No significant decrease in the effectiveness of packaging components that provide shielding, including no increase in radiation levels exceeding the requirements of §71.47 or §71.51(a)(1). Coordinate with the Shielding review.
- No significant decrease in the effectiveness of packaging components that provide criticality control, including no change exceeding the requirements of §71.55(d). (See Section 2.3.5.12.) Coordinate with the Criticality review.
- No change to the contents that significantly affects heat transfer, containment, shielding, or criticality
- No change to the packaging or contents that affects their performance under the tests for hypothetical accident conditions.

The ambient air temperature before and after the tests must remain constant at that value between -29°C (-20°F) and +38°C (100°F) which is most unfavorable for the feature under consideration. The initial internal pressure in the containment vessel must be considered to be the maximum normal operating pressure, unless a lower internal pressure consistent with the selected ambient temperature is less favorable.

2.3.5.1 Heat

Verify that the evaluation for the heat condition is adequate. Confirm that the maximum temperatures used for this evaluation are consistent with the Thermal Evaluation chapter of the SARP. The evaluations should consider the maximum normal operating pressure in combination with the maximum internal heat load and any residual fabrication stresses.

Verify that any differential thermal expansions and possible geometric interferences have been considered.

Verify that the stresses are within the limits for normal condition loads.

2.3.5.2 Cold

Verify that the evaluation for the cold condition is adequate. Confirm that the temperatures used for this evaluation are consistent with the Thermal Evaluation chapter of the SARP. The evaluations should consider the minimum internal pressure with the minimum internal heat load and any residual fabrication stresses. The minimum decay heat should be zero unless the SARP provides a minimum heat load as a condition of package approval.

Verify that differential thermal expansions which could result in possible geometric interferences have been considered. Confirm that possible freezing of liquids and brittle fracture of materials have been considered.

Verify that the stresses are within the limits for normal condition loads.

2.3.5.3 Reduced External Pressure

Ensure that the SARP adequately evaluates the package design for the effects of reduced external pressure equal to 25 kPa (3.5 psi) absolute. Verify that the SARP considers the greatest possible pressure difference between the inside and outside of the package as well as between the inside and outside of the containment system.

2.3.5.4 Increased External Pressure

Determine that the SARP adequately evaluates the package design for the effects of increased external pressure equal to 140 kPa (20 psi) absolute. Verify that the SARP considers this loading condition in combination with minimum internal pressure. Confirm that the SARP considers the greatest possible pressure difference between the inside and outside of the package as well as between the inside and outside of the containment system. Ensure that the SARP has considered the possibility of buckling (see NUREG/CR-4554, Vol. 6).

2.3.5.5 Vibration

Determine that the SARP adequately evaluates the package design for the effects of vibration incident to transport. A fatigue analysis should be provided for highly stressed systems, considering the combined stresses due to vibration, temperature changes, and pressure loads. If closure bolts are reused, verify that the bolt preload is included in the fatigue evaluation. NUREG/CR-6007[16] provides guidance on bolt evaluation. Verify that a resonant vibration condition, which can cause rapid fatigue damage, is not present in any packaging component. The effect on package internals should be considered. Additional guidance for vibration evaluation is provided in NUREG/CR-2146[17] and NUREG/CR-0128.[18]

2.3.5.6 Water Spray

Review the package design for the effects of the water spray test. Verify that this test has no significant effect on material properties.

2.3.5.7 Free Drop

Review the package design for the effects of the free drop test.

Review the evaluation of the closure lid bolt design for the combined effects of free drop impact force, internal pressures, thermal stress, O-ring compression force, and bolt preload. Bolt evaluation methods are presented in NUREG/CR-6007.

Review the evaluation of other package components, such as port covers, port cover plates, and shield enclosures, for the combined effects of package drop impact force, internal pressures, and thermal stress.

2.3.5.8 Corner Drop

Review the package design for the effects of the corner drop test, if applicable.

2.3.5.9 Compression

Review the package design for the effects of the compression test, if applicable.

2.3.5.10 Penetration

Review the evaluation of the package for the penetration test. Verify that the SARP considers the most vulnerable package location.

2.3.5.11 Structural Requirements for Fissile Material Packages

The SARP should demonstrate that there will be no reduction in effectiveness of the packaging, including:

- The geometric form of the contents is not substantially altered.
- The containment system precludes inleakage of water, unless such inleakage has been assumed in the criticality analysis of arrays under normal conditions of transport as specified in §71.59(a)(1).

- The total effective packaging volume on which nuclear criticality safety is assessed is not reduced by more than 5%.
- The effective spacing between fissile contents and the outer surface of the packaging is not reduced by more than 5%.
- No occurrence of an aperture in the outer surface of the packaging is large enough to permit the entry of a 10-cm (4-in.) cube.

Coordinate with the Criticality review as appropriate.

2.3.6 Structural Evaluation for Hypothetical Accident Conditions

The evaluation under hypothetical accident conditions must be based on sequential application of the tests specified in §71.73, in the order indicated, to determine their cumulative effect on a package. The evaluation of the ability of a package to withstand any one test must consider the damage resulting from the preceding tests. In addition, as stated in Section 2.3.5, the tests under normal conditions of transport must not affect the package's ability to withstand the hypothetical accident condition tests.

Verify that the SARP has properly determined the effects of the hypothetical accident condition tests on both the packaging and its contents. The most unfavorable effects of these tests should be identified for evaluation in the Thermal, Containment, Shielding, and Criticality Evaluation chapters of the SARP. Ensure that the SARP has addressed the effects of the tests on the:

- Components required for heat transfer or insulation
- Components of the containment system (plastic deformation of the containment closure system is generally unacceptable)
- Shielding components
- Components required for subcriticality
- Displacement, deformation, and geometry of the contents.

Coordinate with the Thermal, Containment, Shielding, and Criticality reviews as appropriate.

With respect to the initial conditions for the tests (except for the water immersion tests), the ambient air temperature before and after the tests must remain constant at that value between -29°C (-20°F) and +38°C (100°F) which is most unfavorable for the feature under consideration. The initial internal pressure within the containment system must be the maximum normal operating pressure unless a lower internal pressure consistent with the selected ambient temperature is less favorable.

2.3.6.1 Free Drop

Review the evaluation of the free drop test. Verify that structural evaluation has addressed the most unfavorable drop orientation, including cg-over-corner, oblique orientation with secondary impact (slap down), side drop, and drop onto the closure systems. Determination of the most unfavorable orientation must consider the entire sequence of tests, and the most unfavorable

orientation might not be the same for all components. If a feature such as a tie-down component is a structural part of the package, it should be addressed in the evaluation.

For a package with lead shielding, the effects of lead slump should be evaluated. The lead slump determined should be consistent with that used in the shielding evaluation. Lead slump is discussed in NUREG/CR-4554, Vol. 3.

2.3.6.2 Crush

Review the evaluation of the package for the dynamic crush test, if applicable. Verify that the choice of the most unfavorable orientation has been justified.

2.3.6.3 Puncture

Review the evaluation of the package for the puncture test. Verify that the most unfavorable orientation has been identified and justified. Any damage resulting from the free drop and crush tests must be included in the evaluation. Ensure that punctures at oblique angles, near a support, at a valve, and at a penetration or protrusion have been considered, as appropriate. Confirm that the puncture test does not result in peripheral damage that could jeopardize the package during the subsequent thermal and water-immersion tests (e.g., loss of package lid which could result in melting of seals).

Although analytical methods are available for predicting puncture, empirical formulas derived from puncture test results of laminated panels are usually used for design of packages. The Nelm's formula, developed specifically for package design, provides the minimum thickness needed for preventing the puncture of the steel surface layer of a typical steel-lead-steel laminated cask wall. A description of methods for puncture evaluation is provided in NUREG/CR-4554, Vol. 7. Additional considerations for puncture testing are identified in NRC Bulletin 97-02.[19]

2.3.6.4 Thermal

Coordinate with the Thermal review to verify that the structural design is evaluated for the effects of a fully engulfing fire, as specified in §71.73(c)(4). Any damage resulting from the free drop, crush, and puncture conditions must be incorporated into the initial condition of the package for the fire test. Determination of the maximum pressure in the package during or after the test must consider the temperatures resulting from the fire and any increase in gas inventory caused by combustion or decomposition processes. Verify that the maximum thermal stresses, which can occur either during or after the fire, are properly evaluated and are consistent with the Thermal Evaluation chapter of the SARP.

2.3.6.5 Immersion—Fissile Material

If the contents include fissile material subject to the requirements of §71.55, and if water inleakage has not been assumed for the criticality analysis, review the evaluation of the test of a damaged specimen immersed under a head of water of at least 0.9 m (3 ft.) in the attitude for which maximum leakage is expected.

2.3.6.6 Immersion—All Packages

Review the evaluation of a separate, undamaged specimen subjected to water pressure equivalent to immersion under a head of water of at least 15 m (50 ft.). For test purposes, an external pressure of water of 150 kPa (21.7 psi) gauge is considered to meet these conditions.

2.3.7 Lifting and Tie-Down Standards for All Packages

2.3.7.1 Lifting Devices

Review the design and evaluation of lifting devices that are a structural part of the package, their connection to the package body, and the package body in the local area around the lifting devices. Verify that the evaluation demonstrates these devices comply with the requirements of §71.45(a), including failure under excessive load.

2.3.7.2 Tie-Down Devices

Review the design and evaluation of tie-down devices that are a structural part of the package, their connection to the package body, and the package body in the local area around the tie-down devices. Verify that the evaluation demonstrates these devices comply with the requirements of §71.45(b), including failure under excessive load.

2.3.8 Structural Evaluation of Special Pressure Conditions

2.3.8.1 Special Requirement for Irradiated Nuclear Fuel

Verify that packages for irradiated nuclear fuel with an activity greater than 37 PBq (10^6 Ci) are appropriately evaluated to demonstrate that their containment system can withstand an external water pressure of 2 MPa (290 psi) for a period of at least one hour without collapse, buckling, or inleakage of water. This pressure should be applied directly to the containment system, and no structural support from other package components should be considered.[20] Ensure that the stresses in the vicinity of the closure regions do not result in permanent deformation.

2.3.8.2 Analysis of Pressure Test

As required by §71.85(b), prior to first use of each packaging with a maximum normal operating pressure exceeding 35 kPa (5 psi) gauge, the containment system must be pressure tested at 150% of its maximum normal operating pressure. A similar test (125% of the design pressure) is prescribed by Section III of the B&PV Code. If such tests are applicable, confirm that analysis in the SARP demonstrates that they can be performed safely.

2.3.9 Appendix

Confirm that the appendix includes a list of references, copies of applicable references if not generally available to the reviewer, computer code descriptions, input and output files, test results, and other appropriate supplemental information.

2.4 Evaluation Findings

2.4.1 Findings

The review should ensure that the information presented supports a conclusion that the regulatory requirements in Section 2.2 above are satisfied.

The TRR should include a finding similar to the following:

> Based on review of the statements and representations in the SARP, the staff concludes that the structural design has been adequately described and evaluated and that the package design meets the structural requirements of 10 CFR Part 71.

2.4.2 Conditions of Approval

The TRR should clearly identify any conditions of approval that should be included in Section 5 of the certificate of compliance. In addition to specifications of authorized contents and information specified on the engineering drawings, conditions of approval typically applicable to the Structural Evaluation chapter of the SARP include:

- Maximum weight of the package (if not indicated on drawings); minimum weight, if applicable
- Maximum weight of the contents, including shoring, packing materials, and other components not defined as part of the packaging (if not indicated on drawings); minimum weight, if applicable.

2.5 References

1. U.S. Nuclear Regulatory Commission, "Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material," Regulatory Guide 7.8.
2. U.S. Nuclear Regulatory Commission, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels," Regulatory Guide 7.6.
3. American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code*, 1998.
4. U.S. Nuclear Regulatory Commission, "SCANS (Shipping Cask ANalysis System): A Microcomputer Based Analysis System for Shipping Cask Design Review," NUREG/CR-4554 (UCID-20674), February 1990.
5. U.S. Nuclear Regulatory Commission, "Buckling Analysis of Spent Fuel Basket," NUREG/CR-6322 (UCRL-ID-119697), May 1995.
6. Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More," ANSI N14.6, September 1986.
7. U.S. Nuclear Regulatory Commission, "Control of Heavy Loads at Power Plants," NUREG-0612, July 1980.
8. U.S. Nuclear Regulatory Commission, "Fabrication Criteria for Shipping Containers," NUREG/CR-3854 (UCRL-53544), March 1985.

9. U.S. Nuclear Regulatory Commission, "Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials," NUREG/CR-3019 (UCRL-53044), March 1985.

10. U.S. Nuclear Regulatory Commission, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m)," Regulatory Guide 7.11.

11. U.S. Nuclear Regulatory Commission, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater than 4 Inches (0.1 m)," Regulatory Guide 7.12.

12. U.S. Nuclear Regulatory Commission, "Hydrogen Gas Ignition during Closure Welding of a VSC-24 Multi-Assembly Sealed Basket," NRC Information Notice 96-34, May 31, 1996.

13. G. C. Mok, et al. "Guidelines for Conducting Impact Tests of Shipping Containers for Radioactive Material," UCRL-ID-121673, Lawrence Livermore National Laboratory, September 1995.

14 . Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (1985 Edition), Third Edition (as amended 1990), International Atomic Energy Agency, Vienna, 1990.

15. U.S. Nuclear Regulatory Commission, "Methods for Impact Analysis of Shipping Containers," NUREG/CR-3966 (UCID-20639), November 1987.

16. U.S. Nuclear Regulatory Commission, "Stress Analysis of Closure Bolts for Shipping Casks," NUREG/CR-6007 (UCRL-ID-110637), January 1993.

17. U.S. Nuclear Regulatory Commission, "Dynamic Analysis to Establish Normal Shock and Vibration of Radioactive Material Shipping Packages, Volume 3: Final Summary Report," NUREG/CR-2146, Vol. 3, October 1983.

18. U.S. Nuclear Regulatory Commission, "Shock and Vibration Environments for a Large Shipping Container During Truck Transport (Part II)," NUREG/CR-0128, August 1978.

19. U.S. Nuclear Regulatory Commission Bulletin 97-02, "Puncture Testing of Shipping Packages under 10 CFR Part 71," September 23, 1997.

20. U.S. Nuclear Regulatory Commission, "Compatibility with the International Atomic Energy Agency," *Federal Register*, Volume 60, No. 188, September 28, 1995, p. 50257.

3 THERMAL REVIEW

This review verifies that the thermal performance of the package design has been adequately evaluated for the tests specified under normal conditions of transport and hypothetical accident conditions and that the package design meets the thermal requirements of 10 CFR Part 71.

The Thermal review is based in part on the descriptions and evaluations presented in the General Information and Structural Evaluation chapters of the SARP. Similarly, results of the Thermal review are considered in the Structural review and in the review of subsequent chapters of the SARP. An example of information flow for the Thermal review is shown in Figure 3-1.

Although 10 CFR Part 71 specifies only a few explicit thermal requirements for packages (e.g., maximum allowable surface temperature), the thermal performance of the package under normal conditions of transport and hypothetical accident conditions must be addressed in the structural evaluation, and the combined structural/thermal performance of the package affects its ability to meet the containment, shielding, and subcriticality requirements of the regulation. Consequently, the Thermal review focuses on confirming the SARP evaluation of the effects of these tests and on coordinating these effects with the review of the Structural Evaluation, Containment, Shielding Evaluation, and Criticality Evaluation chapters.

3.1 Areas of Review

The description and evaluation of the package thermal design should be reviewed. The Thermal review should include the following:

3.1.1 Description of Thermal Design

- Design Features
- Decay Heat of Contents
- Codes and Standards
- Summary Tables of Temperatures
- Summary Table of Maximum Pressures

3.1.2 Material Properties, Thermal Limits, and Component Specifications

- Material Properties
- Temperature Limits
- Component Specifications

3.1.3 General Considerations for Thermal Evaluations

- Evaluation by Analysis
- Evaluation by Test

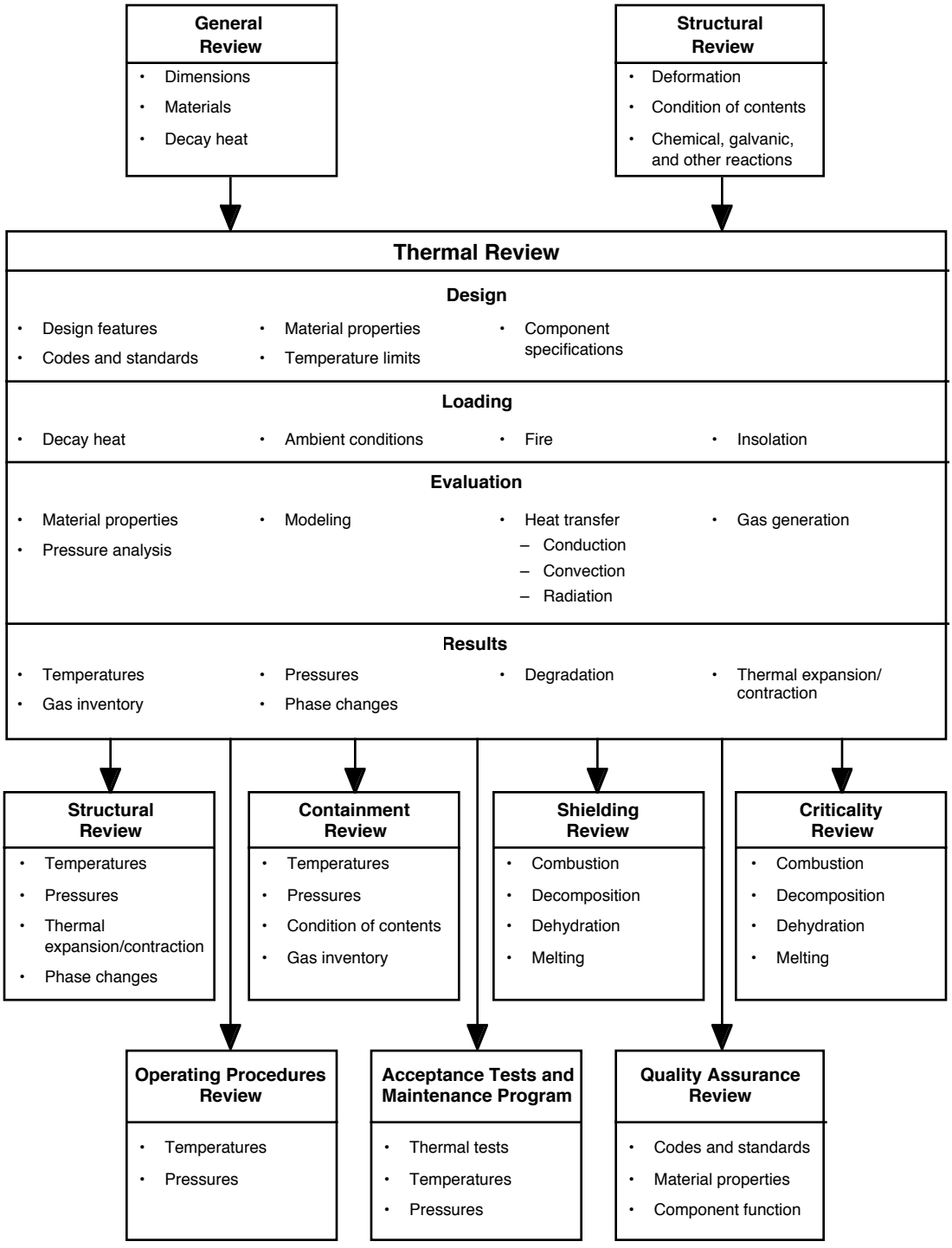

Figure 3-1 Example of Information Flow for the Thermal Review

3.1.4 Thermal Evaluation under Normal Conditions of Transport

- Initial Conditions
- Effects of Tests
- Maximum Normal Operating Pressure
- Maximum Thermal Stresses

3.1.5 Thermal Evaluation under Hypothetical Accident Conditions

- Initial Conditions
- Effects of Thermal Tests
- Maximum Temperatures and Pressures
- Maximum Thermal Stresses

3.1.6 Thermal Evaluation of Maximum Accessible Surface Temperature

3.1.7 Appendix

- Description of Test Facilities and Equipment
- Test Results
- Applicable Supporting Documents or Specifications
- Analyses Details

3.2 Regulatory Requirements

Regulatory requirements of 10 CFR Part 71 applicable to the thermal evaluation are as follows:

- The package design must be described and evaluated to demonstrate that it satisfies the thermal requirements of 10 CFR Part 71. [§71.31(a)(1), §71.31(a)(2), §71.33, §71.35(a)]
- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]
- The package must be made of materials of construction that assure there will be no significant chemical, galvanic, or other reactions, including reactions due to possible inleakage of water, among the packaging components, among package contents, or between the packaging components and the package. The effects of radiation on the materials of construction must be considered. [§71.43(d)]
- The performance of the package must be evaluated under the tests specified in §71.71 for normal conditions of transport. [§71.41(a)]
- The package must be designed, constructed, and prepared for shipment so there would be no loss or dispersal of contents, no significant increase in external surface radiation

levels, and no substantial reduction in the effectiveness of the packaging under the tests specified in §71.71 for normal conditions of transport. [§71.43(f), §71.51(a)(1)]

- The package must be designed, constructed, and prepared for transport so that in still air at 38°C (100°F) and in the shade the accessible surface temperature does not exceed 50°C (122°F) in a nonexclusive-use shipment or 85°C (185°F) in an exclusive-use shipment. [§71.43(g)]
- The performance of the package must be evaluated under the tests specified in §71.73 for hypothetical accident conditions. [§71.41(a)]
- The package design must not rely on mechanical cooling systems to meet containment requirements. [§71.51(c)]

3.3 Review Procedures

The following procedures are generally applicable to the review of the Thermal Evaluation chapter of the SARP. These procedures correspond to the Areas of Review listed in Section 3.1 of this PRG.

3.3.1 Description of Thermal Design

3.3.1.1 Design Features

Review the thermal design features presented in the General Information and Thermal Evaluation chapters of the SARP, including:

- Structural and mechanical means for the transfer of heat (e.g., fill gas, baskets or other internal supporting structures, physical contacts between components, coolant receptacles, type and volume of coolants, cooling fins, and surface conditions of the packaging components)
- Insulating features, including gaps and insulating materials
- Configuration and materials of the contents.

Information on design features should include location, dimensions, tolerances, materials, and other data as appropriate.

Confirm that the text and sketches describing the thermal design features are consistent with the engineering drawings.

3.3.1.2 Decay Heat of Contents

Verify that the maximum decay heat is consistent with that described in the General Information chapter of the SARP, with the radioactivity of the contents, and with the source terms used in the Shielding Evaluation chapter. Coordinate as appropriate with the Shielding review.

Minimum decay heat is discussed in Section 3.3.2.2 below.

3.3.1.3 Codes and Standards

Verify that any codes or standards applicable to the thermal design of the package are identified and appropriate, including those for material specifications and fabrication. Ensure that such codes and standards are consistent with those specified in the General Information and Structural Evaluation chapters of the SARP. Determine if these codes or standards specify temperature limits for materials.

3.3.1.4 Summary Tables of Temperatures

Review the tables that summarize the maximum temperatures of all materials and components affecting structural integrity, thermal performance, containment, shielding, and criticality. As a minimum, these tables should include:

- The maximum temperatures under normal conditions of transport
- The maximum temperatures under hypothetical accident conditions, and the time after initiation of the fire at which they occur
- The maximum temperatures for the post-fire steady-state condition.

Confirm that these temperatures are consistent with those of the General Information, Structural Evaluation, and Containment chapters.

Minimum package temperatures are discussed in Section 3.3.2.2 below. In general, the minimum temperature of all materials and components will be -40°C (-40°F).

3.3.1.5 Summary Table of Maximum Pressures

Verify that a summary table includes the maximum normal operating pressure and the maximum pressure in the containment system(s) under hypothetical accident conditions. Determine if other confined volumes of the package are subject to maximum pressure limitations (e.g., outer shell, neutron shielding system, contents) and that such limitations are included in the table as appropriate. Confirm that these pressures are consistent with those in the General Information, Structural Evaluation, and Containment chapters.

3.3.2 Material Properties, Temperature Limits, and Component Specifications

3.3.2.1 Material Properties

Verify that appropriate properties are specified for materials which affect heat transfer through the package to (or from) the environment, pressures in the package, and thermal stresses. Material properties and the temperature range over which they are designated should be consistent with those used in the structural and thermal evaluations. If a property is specified as temperature independent, ensure that its value is conservative compared with a temperature-dependent specification. Note that a conservative value for heat removal under normal conditions of transport is not necessarily conservative for the thermal test under hypothetical accident conditions. The SARP should provide an authoritative reference for each material property. In general, textbooks are not acceptable references. If the applicant determines thermal properties experimentally, the experiments should be conducted under his quality assurance program, and the adequacy of the experiments should be reviewed.

Properties of package (packaging and contents) materials that may be applicable to the heat-transfer evaluation include density, thermal conductivity, specific heat, viscosity, emissivity, and absorptivity. Confirm that the absorptivities and emissivities are appropriate for the package surface conditions, geometries, and radiant spectra. If the SARP justifies an absorptivity less than unity for insolation based on external packaging surface conditions, ensure that controls and procedures are in place to maintain these conditions during service life. Coordinate with the Operating Procedures review as applicable.

Properties of package material that affect thermally-induced pressures or stresses may include the coefficient of thermal expansion, modulus of elasticity, and Poisson's ratio. Verify that these properties are consistent with those in the Structural Evaluation chapter, as applicable.

If materials undergo chemical or physical changes (e.g., decomposition, dehydration, or combustion), verify that the temperatures at which these conditions occur are presented and that the corresponding material properties (e.g., conductivity, specific heat, density) are appropriate prior to and following the change.

3.3.2.2 Temperature Limits

Confirm that the maximum allowable temperatures are specified for each package material or component, as appropriate. If applicable, ensure that the SARP distinguishes between steady-state and short-term temperature limits.

For spent fuel, the SARP should justify the allowable fuel/cladding temperatures. This justification should consider fuel/cladding materials, irradiation conditions, transport environment (including the package fill gas), temperature history of the fuel since removal from the reactor, and intended post-transport storage or disposition. Temperature limits should address creep, creep rupture, diffusion controlled cavity growth, eutectic melting, and other conditions as appropriate.

The minimum temperature of all materials and components will generally be that of the ambient environment, and the minimum allowable temperatures should not exceed -40°C (-40°F) for the conditions of §71.71(c)(2) and -29°C (-20°F) for the other tests of §71.71 and §71.73.

Ensure that the temperatures listed in the summary tables are within the allowable temperature limits.

3.3.2.3 Component Specifications

Ensure that technical specifications are provided for package components (e.g., pressure relief valves, fusible plugs, valves, seals), as appropriate. Confirm that temperature and pressure specifications are not exceeded. Verify that appropriate specifications (e.g., rupture pressure) are included on the engineering drawings.

3.3.3 General Considerations for Thermal Evaluations

Thermal evaluations of the package design can be performed by analysis, test, or a combination of both methods. The evaluations should demonstrate that the thermal performance of the package meets the criteria discussed in Section 3.3.4 for normal conditions of transport and

Section 3.3.5 for hypothetical accident conditions. The review of these evaluations should verify that:

- The most unfavorable initial regulatory conditions have been addressed. RG 7.8 provides guidance on selection of initial conditions. Note that the thermal evaluations should consider a package that has first been subjected to the structural tests under normal conditions of transport and hypothetical accident conditions, as appropriate. Coordinate with the Structural review.
- The most unfavorable orientations have been considered. The most unfavorable orientation for one component may not be the most unfavorable for another component.
- All regulatory test requirements have been included in the evaluation.
- The evaluation methods are appropriate for the thermal conditions considered and follow accepted practices and precepts.
- The time interval after the fire test is adequate to assure that maximum component temperatures and post-fire steady-state temperatures have been determined.
- The results are interpreted correctly.
- The thermal evaluations appropriately address pass/fail criteria and the design margins for package temperatures, pressures, and thermal stresses. Verify that these discussions include the effects of uncertainties in thermal properties, modeling, analytical methods, test conditions, and diagnostics, as appropriate.

3.3.3.1 Evaluation by Test

If the package is evaluated by test, the review should include the following:

- Verify that the test facility and instrumentation are adequately described and that the test methods and equipment are sufficient for determining the thermal response of the package. Section 3.3.6.1 provides additional detail on the type of information appropriate.
- Verify that the test procedures, test conditions, and test results are adequately documented. Section 3.3.6.2 provides additional detail on test documentation.
- Verify that the test specimen has been fabricated using the materials, methods, and quality assurance specified for the package design. Any differences should be identified and the effects evaluated in the SARP. The test specimen should include all components that could affect the test results. Substitutes for the contents or other simulated components should have the same weight, thermal properties, and interaction with the packaging as the actual contents. Thermal testing of reduced-scale packages should generally be avoided. If scale models are used, the SARP should justify that the evaluation is applicable to the actual package design.
- Verify that decay heat of the contents is properly addressed in the tests or is otherwise included in post-test analysis of the results.

- Verify that all test results are evaluated and their implications correctly interpreted. Unexpected or unexplainable test results indicating possible testing problems or non-reproducible thermal performance should be described and evaluated.
- Verify that the interpretation of the test results addresses differences between test conditions and regulatory conditions. For example, decay heat and regulatory ambient temperature and insolation can result in package temperatures that differ from those of the tested package. Such test results may need to be extended to the regulatory conditions by detailed analysis.
- Review the video and photographs of the tests as appropriate.
- Verify that the test results are reliable and repeatable. Test results should convincingly show that any package fabricated in accordance with the approved design will meet regulatory requirements.
- Review the criteria for evaluating pass/fail for the test conditions. Compare the test results with these criteria. If acceptance tests are performed after the thermal testing, the acceptance tests should be performed according to appropriate codes and standards.

Additional guidance on thermal testing of packages is provided in UCRL-ID-110445.[1]

3.3.3.2 Evaluation by Analysis

If the package is evaluated by analysis, the review should include the following:

- Verify that the SARP clearly describes the analysis methods and models, and that they are appropriate for the thermal conditions considered.
- Verify that the initial and boundary conditions are appropriate.
- Verify that all assumptions, including those in modeling heat sources and heat transfer paths and modes, are clearly stated and justified.
- Verify that appropriate expressions are used for conductive, convective, and radiative heat transfer among package components and from the surfaces of the package to (and from) the environment.
- Verify that appropriate thermal properties for the package materials are correctly incorporated into the analysis.
- Verify that the computer codes, if applicable, are properly used, benchmarked, and maintained under an appropriate quality assurance program. At least one representative input and output file (or key section of the file) should generally be included in the SARP.
- Verify that the results are correctly interpreted and demonstrate adequate margin of safety based on uncertainties and assumptions of the analysis.
- Review the criteria for evaluating pass/fail for the analysis results. Compare these results with the criteria. The maximum temperatures should be compared to corresponding design-code allowables.

3.3.4 Thermal Evaluation under Normal Conditions of Transport

The package must be evaluated for the effects of the tests in §71.71 on the thermal performance of the package. A description of these tests is presented in Section 2.3.5 of this PRG.

3.3.4.1 Initial Conditions

Except as noted in the next paragraph, the initial conditions for tests under normal conditions of transport must be based on an ambient temperature preceding and following the tests remaining constant at that value between -29°C (-20°F) and 38°C (100°F) which is most unfavorable for the feature under consideration. The initial pressure in the containment system must be considered to be the maximum normal operating pressure unless a lower internal pressure consistent with the ambient temperature is more unfavorable. Note that the determination of maximum normal operating pressure must assume that the package is subjected to the insolation specified in §71.71(c)(1).

As specified in §71.71(c)(2), the effects of low temperature (cold) on the package must consider an ambient temperature of -40°C (-40°F) in still air and shade (no insolation).

3.3.4.2 Effects of Tests

Confirm that the thermal evaluation demonstrates that the tests for normal conditions of transport do not result in significant reduction in package effectiveness, including:

- Significant degradation of the heat-transfer capability (e.g., creation of new gaps between components) or significant degradation of insulating materials.
- Changes in material conditions or properties (e.g., expansion, contraction, thermal stresses, gas generation, and chemical, galvanic, or other reactions) that significantly affect the structural performance of the package. Coordinate with the Structural review.
- Changes in the packaging or contents that significantly affect containment, shielding, or criticality (e.g., thermal decomposition or phase changes of materials). Coordinate with the Containment, Shielding, and Criticality review as appropriate.
- Ability of the packaging to withstand the tests under hypothetical accident conditions. Coordinate also with the Structural review.

3.3.4.3 Maximum and Minimum Temperatures

Verify that the maximum and minimum temperatures of package components and materials under normal conditions of transport are properly evaluated and are consistent with those presented in the summary tables discussed in Section 3.3.1.3 above.

3.3.4.4 Maximum Normal Operating Pressure

Verify that the maximum normal operating pressure is properly evaluated and is consistent with that presented in the summary table discussed in Section 3.3.1.4 above. Maximum normal operating pressure is the maximum gauge pressure that would develop in the containment system in a period of one year under the heat condition of §71.71(c)(1), in the absence of venting, external cooling by an ancillary system, or operational controls during shipment. The evaluation

should include the effects of the appropriate local temperatures and total gas inventory within the containment system. Ensure that the evaluation considers all possible sources of gases within any confined volume, such as:

- Package fill gas
- Saturated vapor, including water vapor from the contents or packaging
- Helium from the radioactive decay of the contents
- Fill gas and fission product gas from spent fuel rods, including a justification for the leakage assumed (see NUREG/CR-6487[2])
- Hydrogen or other gases resulting from thermal or radiolytic decomposition of materials (e.g., water, plastics) or other reactions as appropriate.

Ensure that the SARP demonstrates that hydrogen and other flammable gases comprise less than 5% by volume of the total gas inventory within any confined volume, or otherwise addresses concerns for deflagration of such gases. For spent fuel, the release of fill gas from the fuel rods should not be considered for diluting the hydrogen concentration. Ensure that any operational controls (e.g., reduced shipment time) used to limit hydrogen production are adequate and are appropriately addressed in the Operating Procedures chapter. Note that operational controls during shipment may not be used to limit the maximum normal operating pressure.

If other confined volumes of the package are subject to pressure limitations (e.g., secondary containment, outer shell, neutron shielding system, contents), confirm that pressures within these volumes are appropriately evaluated.

Ensure that these pressures are consistent with those in the General Information, Structural Evaluation, and Containment chapters.

3.3.4.5 Maximum Thermal Stresses

Ensure that the evaluation determines thermal stresses caused by geometric constraints, temperature gradients, and other differential thermal expansions. The evaluation should include the maximum stresses as well as cyclic stresses during the service life of the package. Coordinate with the Structural review.

3.3.5 Thermal Evaluation under Hypothetical Accident Conditions

The package must be evaluated for the effects of the tests in §71.73 on the thermal performance of the package.

3.3.5.1 Initial Conditions

Prior to the fire test, the package design must be evaluated for the effects of the drop, crush (if applicable), and puncture tests. Ensure that the initial physical condition of the package design used in the thermal evaluations considers the most unfavorable effects of these tests. Note that the most unfavorable condition for the fire test is not necessarily the most overall structural damage of the package. Coordinate with the Structural review.

Verify that initial conditions of ambient temperature and internal pressure in the containment system are consistent with the requirements of §71.73(b). Although 10 CFR Part 71 does not specifically address insolation required for the thermal test, supplemental information[3] published with the 1996 rule stated that insolation may be neglected prior to and during the thermal test but should be considered in subsequent package evaluation after the fire. Neglecting insolation prior to the fire will result in an initial temperature in the containment system that is inconsistent with that corresponding to the maximum normal operating pressure and may result in peak temperatures during the fire that are less than those under normal conditions of transport. Consequently, for simplicity and conservatism, the SARP evaluation may frequently include insolation as an initial condition for the fire test.

3.3.5.2 Effects of Thermal Test

Verify that the package design is evaluated for the effects of a fully engulfing fire, as specified in §71.73(c)(4). Ensure that temperature, heat-transfer boundary conditions (including fire-enhanced convection), and an appropriate supply of oxygen are maintained for at least 30 minutes.

Confirm that after the fire:

- No artificial cooling is applied to the package
- The package is subjected to full insolation
- An adequate supply of oxygen is maintained
- All combustion is allowed to proceed until it terminates naturally.

Additional guidance on thermal evaluation of packages is provided in UCRL-ID-110445.

Ensure that the physical condition of the package is clearly identified and appropriately considered in the Containment, Shielding Evaluation, and Criticality Evaluation chapter of the SARP. Coordinate with those reviews as appropriate. In addition, if the package is subjected to the water immersion test of §71.73(c)(5), coordinate with the Structural review to ensure that the post-fire condition of the package has been appropriately addressed.

3.3.5.3 Maximum Temperatures and Pressures

Verify that the evaluation appropriately determines the peak transient temperatures of package components as a function of time after the fire and the maximum temperatures from the post-fire steady-state condition. Ensure that temperatures are corrected for differences between regulatory and test conditions, if applicable. Confirm that these temperatures do not exceed their maximum allowable values. Verify that lead shielding does not reach melting temperature (see Section 5.3.3.2).

Confirm that the evaluation of the maximum pressure in the containment system is based on the maximum normal operating pressure (Section 3.3.4.4) as it is affected by fire-caused increases in package component temperatures. Verify that possible increases in gas inventory resulting from the hypothetical accident condition tests (e.g., from thermal combustion, decomposition, release of fill/fission product gases of spent fuel rods) have been accounted for in the pressure determination.

Ensure that the SARP demonstrates that hydrogen and other flammable gases comprise less than 5% by volume of the total gas inventory within any confined volume, or otherwise addresses concerns for deflagration of such gases, as discussed in Section 3.3.4.4.

If other confined volumes of the package are subject to maximum pressure limitations (e.g., secondary containment, outer shell, neutron shielding system, contents), confirm that pressures in these volumes are appropriately evaluated and are acceptable.

Ensure that these pressures are consistent with those in the General Information, Structural Evaluation, and Containment chapters.

3.3.5.4 Maximum Thermal Stresses

Ensure that the evaluation determines the thermal stresses caused by geometric constraints from temperature gradients and differential thermal expansions. Verify that the maximum thermal stresses, which can occur either during or after the fire, are consistent with those in the Structural Evaluation chapter.

3.3.6 Thermal Evaluation of Maximum Accessible Surface Temperature

Confirm that the maximum temperature of the accessible package surface is less than 50°C (122°F) for a nonexclusive-use shipment or 85°C (185°F) for an exclusive-use shipment when the package is subjected to the heat conditions of §71.43(g). For packages with a significant heat load, coordinate with the Operating Procedures review to ensure that the requirements of §71.87(k) are satisfied.

3.3.7 Appendix

3.3.7.1 Description of Test Facilities and Equipment

Confirm that the descriptions of a test facility include:

- Type of facility (e.g., fire, furnace)
- Method of heating the package (e.g., pool fire, gas burners, electrical heaters)
- Volume and emissivity of the furnace interior
- Types, locations, calibration curves, and measurement uncertainties of all sensors used to measure the fire heat fluxes, fire temperatures, and test package component temperatures and pressures
- The post-fire environment for a time period adequate to attain the post-fire, steady-state condition
- Methods for ensuring an adequate supply and circulation of oxygen for initiating and maintaining the combustion of any burnable package component throughout the fire and post-fire periods until natural termination.

3.3.7.2 Test Reports

Verify that appropriate test reports are included in the appendix. These reports should include:

- Test procedures
- Test package description
- Test initial and boundary conditions
- Test chronologies (planned and actual)
- Photographs of the package components, including any structural or thermal damage, before and after the tests
- Test measurements, including documentation of test package physical changes and temperature and heat-flux histories, as appropriate
- Test results corrected to regulatory conditions
- Methods used to obtain these corrected results.

Confirm that all sensors which measure heat fluxes and temperatures are appropriately positioned and have proper operating ranges for the test conditions. Verify that possible perturbations caused by the presence of these sensors (e.g., by disturbing local convective and radiative heat-transfer conditions) are appropriately considered.

For a pool-fire facility, verify that the fire dimensions and test package relative location conform to the specification in §71.73(c)(4):

- The fire width should extend horizontally between one and four meters beyond any external surface of the package.
- The package should be positioned one meter above the surface of the fuel source.

Since the method of supporting the package in the test facility may locally perturb fire conditions adjoining the test package, verify that such an effect has been appropriately incorporated into the thermal evaluation.

3.3.7.3 Applicable Supporting Documents or Specifications

Verify that appropriate selections from reference documents are included in this appendix. In addition to the documents noted in Sections 3.3.6.1 and 3.3.6.2, these may include a variety of items such as thermal specifications of O-rings and other components, documentation of the thermal properties, computer input and output files, and other appropriate information.

3.3.7.4 Analyses Details

Supplemental calculations may be required to support evaluations presented in the Thermal Evaluation chapter. Verify that all such special analyses are prepared in a manner consistent with Section 3.3.3.2.

3.4 Evaluation Findings

3.4.1 Findings

The reviewer should ensure that the information presented supports a conclusion that the regulatory requirements in Section 3.2 above are satisfied.

The TRR should include a finding similar to the following:

> Based on review of the statements and representations in the SARP, the staff concludes that the thermal design has been adequately described and evaluated, and that the thermal performance of the package meets the thermal requirements of 10 CFR Part 71.

3.4.2 Conditions of Approval

The TRR should clearly identify any conditions of approval that should be included in Section 5 of the certificate of compliance. In addition to specifications of authorized contents and information specified on the engineering drawings, other conditions of approval that may be applicable to the Thermal Evaluation chapter of the SARP include:

- Decay heat limits
- Requirement for exclusive-use shipment due to package surface temperatures.
- Maximum duration of shipment (e.g., to limit hydrogen production).

3.5 References

1. VanSant, J. H., R. W. Carlson, L. E. Fischer, and J. Hovingh, "A Guide for Thermal Testing Transport Packages for Radioactive Material—Hypothetical Accident Conditions," UCRL-ID-110445, Lawrence Livermore National Laboratory, February 9, 1993.
2. U.S. Nuclear Regulatory Commission, "Containment Analysis for Type B Packages Used to Transport Various Contents," NUREG/CR-6487, November 1996.
3. U.S. Nuclear Regulatory Commission, "Compatibility with the International Atomic Energy Agency," *Federal Register*, Volume 60, No. 188, September 28, 1995, p. 50257.

4 CONTAINMENT REVIEW

This review verifies that the package design satisfies the containment requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

The Containment review is based in part on the descriptions and evaluations presented in the General Information, Structural Evaluation, and Thermal Evaluation chapters of the SARP. Similarly, results of the Containment review are considered in the review of Operating Procedures, Acceptance Tests and Maintenance Program, and Quality Assurance. An example of the information flow for the Containment review is shown in Figure 4-1.

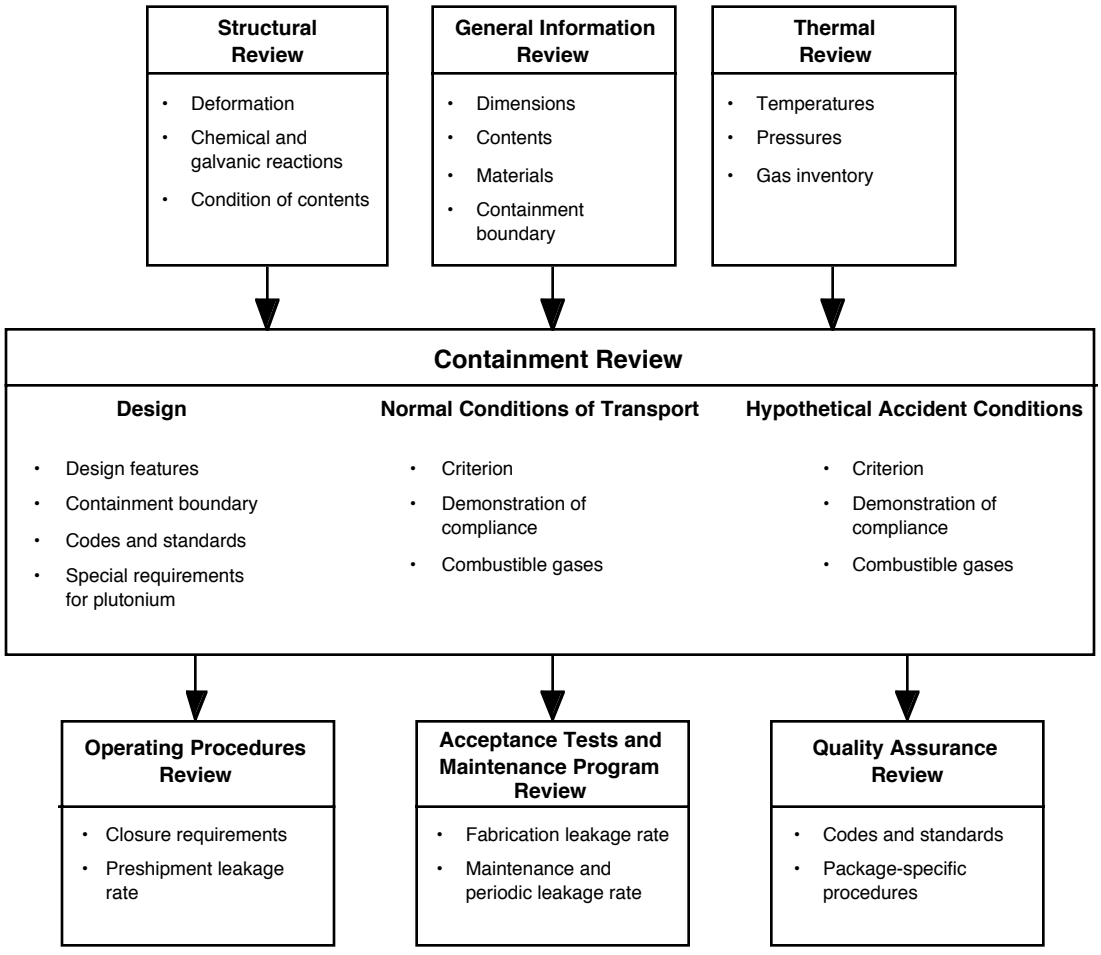

Figure 4-1 Example of Information Flow for the Containment Review

4.1 Areas of Review

The description and evaluation of the containment design should be reviewed. The Containment review should include the following:

4.1.1 Description of Containment Design

- Design Features
- Codes and Standards
- Special Requirements for Plutonium and Spent Fuel

4.1.2 General Considerations for Containment Evaluations

- Fissile Type A Packages
- Type B Packages
- Combustible-Gas Generation

4.1.3 Containment under Normal Conditions of Transport (Type B Packages)

- Containment Design Criterion
- Demonstration of Compliance with Containment Design Criterion

4.1.4 Containment under Hypothetical Accident Conditions (Type B Packages)

- Containment Design Criterion
- Demonstration of Compliance with Containment Design Criterion

4.1.5 Leakage Rate Tests for Type B Packages

4.1.6 Appendix

4.2 Regulatory Requirements

Regulatory requirements of 10 CFR Part 71 applicable to the Containment review are as follows:

- The package design must be described and evaluated to demonstrate that it meets the containment requirements of 10 CFR Part 71. [§71.31(a)(1), §71.31(a)(2), §71.33, §71.35(a)]
- The package must include a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package. [§71.43(c)]
- The package must be made of materials and constructed to assure that there will be no significant chemical, galvanic, or other reactions, including reactions due to possible inleakage of water, among the packaging components, among package contents, or

between the packaging components and the contents. The effects of radiation on the materials of construction must be considered. [§71.43(d)]

- Compliance with the permitted activity release limits for Type B packages may not rely on filters or on a mechanical cooling system. [§71.51(c)]
- The package may not incorporate a feature intended to allow continuous venting during transport. [§71.43(h)]
- Any valve or similar device on the package must be protected against unauthorized operation and, except for a pressure relief valve, must be provided with an enclosure to retain any leakage. [§71.43(e)]
- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]
- A package containing plutonium in excess of 0.74 TBq (20 Ci) must satisfy the special containment requirements for plutonium. [§71.63]
- The maximum activity of radionuclides in a Type A package must not exceed the limits of 10 CFR Part 71, Table A-1. For a mixture of radionuclides, the provisions of Appendix A, paragraph IV apply, except that for krypton-85, an effective A_2 equal to 10 A_2 may be used. [Appendix A, §71.51(b)]
- The package must be designed, constructed, and prepared for shipment to ensure no loss or dispersal of radioactive contents under the tests specified in §71.71 for normal conditions of transport. [§71.43(f)]
- A Type B package must meet the containment requirements of §71.51(a)(1) under the tests specified in §71.71 for normal conditions of transport.
- A Type B package must meet the containment requirements of §71.51(a)(2) under the tests specified in §71.73 for hypothetical accident conditions.

4.3 Review Procedures

The following procedures are generally applicable to the review of the Containment chapter of the SARP. These procedures correspond to the Areas of Review listed in Section 4.1 of this PRG.

4.3.1 Description of the Containment Design

4.3.1.1 Design Features

Review the containment design features presented in the General Information and Containment chapters of the SARP. Design features important to containment include:

- Containment vessel(s)
- Welds

- Seals
- Valves
- Pressure relief devices
- Lids, cover plates, and similar closure devices
- Bolts and bolt torque
- Special containment features for plutonium and spent fuel (see Section 4.3.1.3).

Information on containment design features should include, as appropriate:

- Location, dimensions, and tolerances
- Materials of construction
- Maximum and minimum allowable temperatures of components, including seals
- Maximum and minimum temperatures of components under the tests for normal conditions of transport and hypothetical accident conditions
- Maximum normal operating pressure and maximum pressure in the containment system under hypothetical accident conditions.

The SARP should include a figure or sketch that defines the exact boundary of the containment system. Confirm that all containment boundary penetrations and their method of closure are adequately described. Verify that the containment system is securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package. Coordinate with the Structural and Thermal reviews as appropriate. If penetrations are closed with two seals (e.g., to enable leakage testing), verify which seal is defined as the containment boundary. Ensure that all components of the containment system are shown on the drawings.

Verify that the seal material is appropriate for the package. Ensure that the seal will undergo no galvanic, chemical, or other reaction with the packaging or its contents, will not degrade due to irradiation, and will not be permeable to radioactive gases in the contents. Confirm that the seal grooves are properly sized. Coordinate with the Structural review as appropriate to verify that the specified bolt torque will provide proper seal compression. Cover plates and lids should be recessed or otherwise protected.

Confirm that all containment closure systems can be leakage tested as appropriate. If vent/drain ports or similar penetrations utilize quick-disconnect valves that are not part of the containment boundary, ensure that such valves do not preclude leakage testing of the containment.

Review the maximum and minimum temperatures of all containment system components, including seals, under normal conditions of transport and hypothetical accident conditions. Confirm that the allowable temperature range for each component is not exceeded. Compliance with the containment requirements for Type B packages may not rely on filters or a mechanical cooling system. Coordinate with the Thermal review as appropriate.

Performance specifications for components such as valves and pressure relief devices should be identified, and no device may allow continuous venting. Ensure that the maximum pressure under normal conditions of transport or hypothetical accident conditions does not exceed the specification of pressure relief devices. Coordinate with the Thermal review as appropriate.

Any valve or similar device on the package must be protected against unauthorized operation and, except for a pressure relief valve, must be provided with an enclosure to retain any leakage. (The requirement to provide an enclosure to retain leakage is not intended to require a second containment boundary for Type B packages.)

Confirm that the information regarding the containment system is consistent with that presented in the General Information, Structural Evaluation, and Thermal Evaluation chapters of the SARP.

4.3.1.2 Codes and Standards

Verify that any codes or standards applicable to the containment design of the package are identified and appropriate, including those for material specifications and fabrication. Ensure that such codes and standards are consistent with those specified in the General Information, Structural, and Thermal Evaluation chapters of the SARP. Determine if these codes or standards specify temperature limits for materials.

Evaluation of release rates and performance of leakage testing should be in accordance with ANSI N14.5.[1]

4.3.1.3 Special Requirements for Plutonium and Spent Fuel

If the contents include more than 0.74 TBq (20 Ci) of plutonium, verify that the plutonium is in solid form and that double containment is provided as specified in §71.63(b). Each containment system must separately meet the requirements of §71.51(a)(1) under normal conditions of transport and §71.51(a)(2) under hypothetical accident conditions. Review both containment systems in the same manner.

Containment requirements for spent fuel depend on the condition of the fuel:

- Undamaged fuel is exempt from the double containment requirements of §71.63(b).
- Damaged fuel should be canned in a separate inner container for handling and criticality control. Appropriate material specifications and the design/fabrication criteria for the inner container should be specified, and any credit for the canning in the containment evaluation should be justified. If a screen-type container is used, an appropriate mesh size should be justified. Review the design of the inner container, as applicable.
- Spent fuel in the form of debris, particles, loose pellets, or fragmented rods/assemblies is not considered to be fuel elements in the context of §71.63(b) and requires a separate (inner) containment if the plutonium content exceeds 0.74 TBq (20 Ci).[2] Design, fabrication, and leakage test criteria for the inner container should be the same as those of the outer containment. Review both containment systems as appropriate.

The determination of undamaged fuel should be based, as a minimum, on a review of records to verify that the fuel is undamaged, followed by an visual examination for any obvious damage

prior to loading. For fuel in which reactor records are not available, the level of proof should be evaluated on a case-by-case basis. Coordinate with the Operating Procedures review as appropriate.

4.3.2 General Considerations for Containment Evaluations

4.3.2.1 Fissile Type A Packages

Verify that the contents do not exceed a Type A quantity of radioactive material as specified by Appendix A to 10 CFR Part 71. Note that the only Type A packages subject to 10 CFR Part 71 are fissile-material packages (A-F).

For Type A packages, no loss or dispersal of radioactive material is permitted under normal conditions of transport, as specified in §71.43(f). Although 10 CFR Part 71 does not provide quantitative release limits for containment under hypothetical accident conditions (as it does for Type B packages), the containment must be adequate to ensure subcriticality. Coordinate with the Criticality review as appropriate.

4.3.2.2 Type B Packages

Type B packages must satisfy the quantitative *release* rates of §71.51. ANSI N14.5 provides an acceptable method to determine the maximum permissible volumetric *leakage* rates based on the allowed regulatory *release* rates under both normal conditions of transport and hypothetical accident conditions (L_N and L_A). These two volumetric leakage rates should be converted to maximum allowable *air* leakage rates under reference conditions (temperature, pressures) in accordance with ANSI N14.5. The smaller of L_N and L_A (when converted to reference conditions) is defined as the reference air leakage rate, L_R.

In general, the normal condition leakage rate is the most restrictive. Hence, L_N, when converted to reference conditions, is generally equal to L_R. This situation is assumed in the discussion of containment criteria in Sections 4.3.3 and 4.3.4 below. In the very rare case in which L_R is determined by L_A, the reviewer should refer to ANSI N14.5 to ensure the containment criteria are properly evaluated. Note that this situation can occur only if the releasable source term under hypothetical accident conditions is approximately three orders of magnitude greater than the releasable source term under normal conditions of transport.

The maximum permissible release rate (and leakage rate) for a package that contains different radionuclides is based on an effective A_2, which must be determined according to the provisions of §71.51(b).

Representative analyses for determining simplified containment criteria are provided in NUREG/CR-6487[3] for Type B packages that contain powders, liquids, irradiated fuel rods, gases, or solids. If the SARP uses these analyses, ensure that the assumptions of that document are applicable to the package under consideration. Guidance on containment analyses for aluminum-based spent fuel is provided by WSRC-TR-98-00317.[4]

4.3.2.3 Combustible-Gas Generation

Confirm that the SARP demonstrates that any combustible gases generated in the package during a period of one year do not exceed 5% (by volume) of the free gas volume in any confined

region of the package, or otherwise addresses concerns related to deflagration of such gases. Coordinate with the Structural and Thermal reviews as appropriate.

4.3.3 Containment under Normal Conditions of Transport (Type B Packages)

4.3.3.1 Containment Design Criterion

Confirm that the radionuclides and physical form of the contents evaluated in the Containment chapter are consistent with those presented in the General Information chapter of the SARP. Ensure that the radionuclides include daughter products as appropriate.

Verify that the SARP identifies the constituents which comprise the releasable source term, including radioactive gases, liquids, and powder aerosols. If less than 100% of the contents are considered releasable, evaluate the justification for the lower fraction.

Based on the releasable source term, ensure that the maximum permissible release rate and the maximum permissible leakage rate (L_N) are calculated in accordance with ANSI N14.5. Verify that the maximum normal operating pressure and maximum temperature under normal conditions of transport are consistent with those determined in the Thermal Evaluation chapter of the SARP. Using this pressure and temperature, ensure that the maximum permissible leakage rate L_N is converted to reference cubic centimeters per second (ref cc/s) in accordance with ANSI N14.5.

4.3.3.2 Demonstration of Compliance with Containment Design Criterion

Confirm that the SARP demonstrates that the package meets the containment requirements of §71.51(a)(1) under normal conditions of transport.

If compliance is demonstrated by test:

- Confirm that prior to the test, the leakage rate of the test specimen (when converted to reference conditions) is demonstrated to be less than or equal to L_R, as defined in ANSI N14.5.
- Coordinate with the Structural and Thermal reviews to ensure that a full-scale specimen has been properly tested under the requirements of §71.71. While scale-model testing may yield valuable information for the designer, it is not a reliable or acceptable method for quantifying the leakage rate of a full-scale specimen.
- Verify that the leakage rate of the specimen which has been subjected to the tests of §71.71 does not exceed the maximum allowable leakage rate for normal conditions of transport. To ensure a comparison using consistent units, the leakage rate after the test should generally be converted to reference conditions and then compared with L_R.

If compliance is demonstrated by analysis:

- Confirm that the allowable leakage rate for the fabrication, periodic, and maintenance leakage rate tests is less than or equal to L_R.

- Verify that the structural evaluation shows that the containment system closure region (e.g., bolts, seal, or flange) does not undergo plastic deformation under the tests of §71.71. Coordinate with the Structural review.

4.3.4 Containment under Hypothetical Accident Conditions (Type B Packages)

The review procedures for containment under hypothetical accident conditions are similar to those under normal conditions of transport. Differences relevant to hypothetical accident conditions are noted below.

4.3.4.1 Containment Design Criterion

The releasable source term, maximum permissible release rate, and maximum permissible leakage rate should be based on package conditions and the 10 CFR Part 71 containment requirements under hypothetical accident conditions. Verify that the temperatures, pressure, and physical conditions of the package (including the contents) are consistent with those determined in the Structural Evaluation and Thermal Evaluation chapters of the SARP. Using this pressure and temperature of the contents under hypothetical accident conditions, ensure that the maximum permissible leakage rate L_A is converted to reference cubic centimeters per second (ref cc/s) in accordance with ANSI N14.5.

4.3.4.2 Demonstration of Compliance with Containment Design Criterion

Ensure that the SARP demonstrates that the package satisfies the containment requirements of §71.51(a)(2) under hypothetical accident conditions. Demonstration is similar to that discussed in Section 4.3.3.2, except that the package should be subjected to the tests of §71.73 and the maximum allowable leakage rate at reference conditions must be less than L_A converted to reference conditions.

4.3.5 Leakage Rate Tests for Type B Packages

Using the reference air leakage rate, confirm that the maximum allowable leakage rates for the following tests are determined in accordance with ANSI N14.5:

- Fabrication leakage rate test
- Periodic leakage rate test
- Maintenance leakage rate test
- Pre-shipment leakage rate test.

The fabrication, periodic, and maintenance leakage rate tests should be addressed in the Acceptance Tests and Maintenance Program review. The pre-shipment leakage rate test for assembly verification should be addressed in the Operating Procedures review. Coordinate with those reviews as appropriate.

4.3.6 Appendix

Confirm that the appendix includes a list of references, copies of applicable references if not generally available to the reviewer, test results, and other appropriate supplemental information.

4.4 Evaluation Findings

4.4.1 Appendix

The reviewer should ensure that the information presented supports a conclusion that the regulatory requirements in Section 4.2 above are satisfied.

The TRR should include a finding similar to the following:

> Based on review of the statements and representations in the SARP, the staff concludes that the containment design has been adequately described and evaluated and that the package design meets the containment requirements of 10 CFR Part 71.

4.4.2 Conditions of Approval

The TRR should clearly identify any conditions of approval that should be included in Section 5 of the certificate of compliance. In addition to specifications of authorized contents and information specified on the engineering drawings, other conditions of approval that may be applicable to Containment chapter of the SARP include:

- Requirement to can or provide double containment for damaged fuel
- Maximum duration of shipment (e.g., to limit hydrogen production)
- Other conditions as appropriate.

4.5 References

1. Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment," ANSI N14.5-1997, New York.
2. U.S. Nuclear Regulatory Commission, "Damaged Fuel," ISG-1, Spent Fuel Project Office, 1998.
3. U.S. Nuclear Regulatory Commission, "Containment Analysis for Type B Packages Used to Transport Various Contents," NUREG/CR-6487, November 1996.
4. Westinghouse Savannah River Company, "Bases for Containment Analyses for Transportation of Aluminum-Based Spent Nuclear Fuel," WSRC-TR-98-00317, Aiken, SC, October 1998.

5 SHIELDING REVIEW

This review verifies that the package design meets the external radiation requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

The Shielding review is based in part on the descriptions and evaluations presented in the General Information, Structural Evaluation, and Thermal Evaluation chapters of the SARP. Results of the Shielding review are considered in the review of Operating Procedures, the Acceptance Tests and Maintenance Program, and the Quality Assurance Program. An example of the information flow for the Shielding review is shown in Figure 5-1.

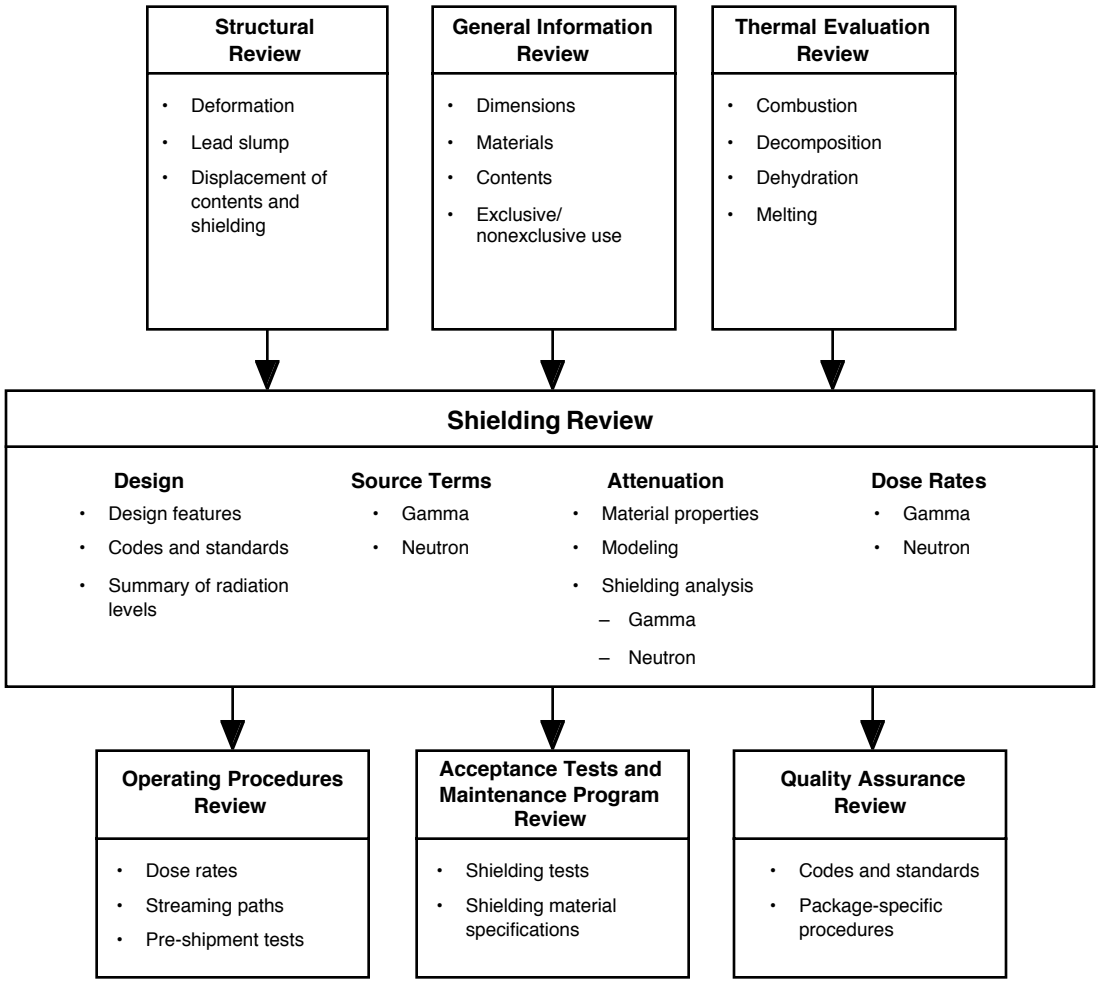

Figure 5-1 Example of Information Flow for the Shielding Review

5.1 Areas of Review

The description and evaluation of the shielding design should be reviewed. The Shielding review should include the following:

5.1.1 Description of Shielding Design

- Design Features
- Codes and Standards
- Summary Table of Maximum Radiation Levels

5.1.2 Radiation Source

- Gamma Source
- Neutron Source

5.1.3 Shielding Model

- Configuration of Source and Shielding
- Material Properties

5.1.4 Shielding Evaluation

- Methods
- Input and Output Data
- Flux-to-Dose-Rate Conversion
- External Radiation Levels

5.1.5 Appendix

5.2 Regulatory Requirements

Regulatory requirements of 10 CFR Part 71 applicable to the shielding review are as follows:

- The package design must be described and evaluated to demonstrate that it meets the shielding requirements of 10 CFR Part 71. [§71.31(a)(1), §71.31(a)(2), §71.33, §71.35(a)]
- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]
- Under the tests specified in §71.71 for normal conditions of transport, the external radiation levels must meet the requirements of §71.47(a) for nonexclusive-use or §71.47(b) for exclusive-use shipments. [§71.47]

- The package must be designed, constructed, and prepared for shipment so that the external radiation levels will not significantly increase under the tests specified in §71.71 for normal conditions of transport. [§71.43(f), §71.51(a)(1)]
- Under the tests specified in §71.73 for hypothetical accident conditions, the external radiation level must not exceed 10 mSv/h (1 rem/h) at one meter from the surface of a Type B package. [§71.51(a)(2)]

5.3 Review Procedures

The following procedures are generally applicable to the review of the Shielding Evaluation chapter of the SARP. These procedures correspond to the Areas of Review listed in Section 5.1 of this PRG.

5.3.1 Description of Shielding Design

5.3.1.1 Design Features

Review the shielding design features presented in the General Information and Shielding Evaluation chapters of the SARP. Design features important to shielding include:

- Location, dimensions, tolerances, and densities of material for neutron or gamma shielding, including those packaging components considered in the shielding evaluation
- Structural components that maintain the integrity of the shielding
- Structural components that maintain the contents in a fixed position within the package
- Heat transfer and insulating features that maintain allowable temperatures of the shielding
- Dimensions of the transport vehicle that are considered in the shielding evaluation, if applicable.

Confirm that the text and sketches describing the shielding design features are consistent with the engineering drawings and the models used in the shielding evaluation.

5.3.1.2 Codes and Standards

Verify that any codes or standards applicable to the shielding design of the package are identified and appropriate, including those for material specifications and fabrication. Ensure that such codes and standards are consistent with those specified in the General Information, Structural, and Thermal Evaluation chapters of the SARP. Determine if these codes or standards specify temperature limits for materials.

Flux-to-dose-rate conversion factors should be consistent with ANSI/ANS6.1.1-1977,[1] as discussed below in Section 5.3.4.3.

5.3.1.3 Summary Table of Maximum Radiation Levels

Review the summary table of maximum radiation levels. Ensure that the maximum levels are presented for both normal conditions of transport and hypothetical accident conditions at the

appropriate locations for nonexclusive or exclusive use (or both), as applicable. Table 5.1 is an example of the information that should be presented for nonexclusive use. A similar table should be presented for exclusive use shipment as appropriate.

Verify that the radiation levels are within the regulatory limits as indicated in Table 5.2. Review the variation of dose rates at different package locations for general consistency. For example, confirm that dose rates decrease as either the distance from the source or as the shielding effectiveness (e.g., thickness) increases.

Table 5.1 Example for Summary Table of External Radiation Levels (Nonexclusive Use)

Normal Conditions of Transport	Package Surface mSv/h (mrem/h)			1 Meter from Package Surface mSv/h (mrem/h)		
Radiation	**Top**	**Side**	**Bottom**	**Top**	**Side**	**Bottom**
Gamma						
Neutron						
Total						
10 CFR 71.47(a) Limit	2 (200)	2 (200)	2 (200)	0.1 (10)*	0.1 (10)*	0.1 (10)*

* Transport index may not exceed 10 for nonexclusive-use shipment.

Hypothetical Accident Conditions*	1 Meter from Package Surface mSv/h (mrem/h)		
Radiation	**Top**	**Side**	**Bottom**
Gamma			
Neutron			
Total			
10 CFR 71.51(a)(2) Limit*	10 (1000)	10 (1000)	10 (1000)

* Applicable to Type B packages only

Table 5.2 Package and Vehicle Radiation Level Limits[a]

Transport Vehicle Use:	**Nonexclusive**	**Exclusive**		
Transport Vehicle Type:	Open or closed	Open (flat-bed)	Open w/ enclosure[b]	Closed
Package (or Freight Container) Limits, mSv/h (mrem/h):				
External surface	2 (200)	2 (200)	10 (1000)	10 (1000)
1 m from external surface	0.1 (10) [d]	No limit		
Roadway or Railway Vehicle (or Freight Container) Limits, mSv/h (mrem/h):				
Any point on the outer surface	N/A	N/A	N/A	2 (200)
Vertical planes projected from outer edges		2 (200)	2 (200)	N/A
Top of . . .		load: 2 (200))	enclosure: 2 (200)	vehicle: 2 (200)
2 m from. . .		vertical planes: 0.1 (10)	vertical planes: 0.1 (10)	outer lateral surfaces: 0.1 (10)
Underside		2 (200)		
Occupied position	N/A[e]	0.02 (2)[f]		

a. The limits in this table are applicable under normal conditions of transport. For Type B packages, the external radiation levels at one meter from the package surface may not exceed 10 mSv/h (1 rem/h) under hypothetical accident conditions. The limits in this table do not apply to excepted packages—see 49 CFR 173.421-426.

b. Securely attached (to vehicle), access-limiting enclosure; package personnel barriers are considered as enclosures.

c. Package secured within vehicle so that its position remains fixed during transportation; no loading or unloading operations between beginning and end of transportation. Otherwise limit is 2 mSv/h (200 mrem/h).

d. Transport index may not exceed 10 for nonexclusive-use shipment.

e. No dose limit is specified, but separation distances apply to packages with Radioactive Yellow-II or Radioactive Yellow-III labels–see 49 CFR 177.842(b).

f. Does not apply to private carriers if exposed personnel under their control wear dosimetry devices in conformance with 10 CFR 20.1502.

5.3.2 Radiation Source

Confirm that the contents used in the shielding evaluation are consistent with those specified in the General Information chapter of the SARP. If the package is designed for multiple types of contents, ensure that the contents producing the highest external dose rate at each location are clearly identified and evaluated.

If the contents include spent fuel, verify that limitations on burnup, enrichment, and cooling time have been properly addressed. Although the maximum fuel enrichment is important for criticality analysis, the neutron source term for shielding evaluations can increase significantly with decreasing initial enrichment (for constant burnup and cooling time). Ensure that the SARP specifies a minimum initial enrichment for the fuel as appropriate. Verify that the cross sections used to calculate the source terms are applicable for the burnup indicated; some cross-section libraries are not valid for higher burnup.

5.3.2.1 Gamma Source

Review the method used to determine the gamma source term. Ensure that the source contribution from radioactive daughter products is included if it produces higher dose rates than the contents without decay. If the radioactive nuclides and gamma spectra are calculated with a computer code, review the key parameters described in the SARP or listed in the input file. Verify that the production of secondary gammas (e.g., from (n,γ) reactions in shielding material or bremsstrahlung from beta decay) is either calculated as part of the shielding evaluation (see Section 5.3.4.1) or otherwise appropriately included in the source term.

If the contents include spent fuel, verify that the gamma source terms are determined for both the spent fuel and activated hardware. If the package is intended to transport other hardware such as control assemblies or shrouds, ensure that the source terms from these components are also included if applicable. Note whether the source terms are specified per fuel rod, per assembly, per total assemblies, or per metric ton, and ensure that the total source is correctly used in the shielding evaluation.

Confirm that the results of the source term determination are presented as a listing of gammas per second, or MeV per second, as a function of energy. The activity (or mass) of each nuclide that contributes significantly to the source term should also be provided as supporting information.

5.3.2.2 Neutron Source

Review the method used to determine the neutron source term. Verify that the method considers, as appropriate, neutrons from both spontaneous fission and from (α,n) reactions. If the SARP assumes that either of these source contributions is negligible, ensure that an appropriate justification is provided. Verify that the production of neutrons from subcritical multiplication is either calculated as part of the shielding evaluation (see Section 5.3.4.1) or otherwise appropriately included in the source term.

Confirm that the results of the source term calculation, if applicable, are presented as a listing of neutrons per second as a function of energy. The contributions from spontaneous fission and (α,n) should be separately identified. The activity (or mass) of each nuclide that contributes significantly to the source terms should also be provided as supporting information.

5.3.3 Shielding Model

Review the Structural and Thermal Evaluation chapters of the SARP to determine the effects that the tests for normal conditions of transport and hypothetical accident conditions have on the packaging and its contents. Verify that the models used in the shielding calculation are consistent with these effects and with the engineering drawings. Coordinate with the Structural and Thermal reviews as appropriate.

5.3.3.1 Configuration of Source and Shielding

Verify the dimensions of the source and packaging used in the shielding models, and ensure that tolerances have been appropriately considered. If contents can be positioned at varying locations or with varying densities, ensure that the location and physical properties of the contents used in the evaluation are those resulting in the maximum external radiation levels. For example, the source configuration that maximizes the radiation level on the side of the package might not be the same source configuration that maximizes the radiation level on the top or bottom. Ensure that any changes in configuration (e.g., displacement of source or shielding, reduction in shielding) resulting under normal conditions of transport or hypothetical accident conditions have been included, as appropriate.

For spent fuel, confirm that the spent-fuel region and activated-hardware regions (e.g., top/bottom end-pieces and plenum) are properly located in the model. Verify that flux peaking, both radially and axially within the fuel, has been treated appropriately.

In general, the shielding model and evaluation need address radiation levels from only one package and show that the requirements of §71.47 are satisfied. Based on external radiation levels measured prior to shipment, multiple packages may be combined in conveyance in accordance with 49 CFR 177.842 (nonexclusive use), 49 CFR 173.441 (exclusive use), and other applicable DOT regulations. (Combining packages with fissile material must also address criticality-safety restrictions, as discussed in Section 6 of this PRG.)

For exclusive-use shipments in which the analysis is based on the radiation levels of §71.47(b), confirm that dimensions of the transport vehicle and package location are included as appropriate. These dimensions or vehicle type, as well as positioning of the packages, become limiting conditions in the certificate of compliance if used in the evaluation. For some packages, the use of radiation levels at distances from the package surface instead of the vehicle surface may be sufficient to demonstrate compliance without the need to specify vehicle dimensions.

Verify that the dose point locations in the shielding model include all locations prescribed in §§71.47(a) or 71.47(b), and §71.51(a)(2) as appropriate. Ensure that these points are chosen to identify the location of the maximum radiation levels. Confirm that voids, streaming paths, and irregular geometries are included in the model or otherwise treated in an adequate manner. For exclusive-use shipments, ensure that the determination of the radiation levels on the bottom surface of the vehicle, at 2 m from the vehicle, and in normally occupied positions account for the contribution from ground scatter, as appropriate.

5.3.3.2 Material Properties

Verify the appropriate material properties (e.g., mass densities and atom densities) used in the shielding models of the packaging, contents, and conveyance (if applicable). For uncommon

materials, especially foams, plastics, and other hydrocarbons, the source of data should be referenced. Material specifications should be consistent with those in the engineering drawings. Confirm that shielding properties will not degrade significantly during the service life of the packaging (e.g., degradation of foam or dehydration of hydrogenous materials).

Ensure that any changes resulting under normal conditions of transport or hypothetical accident conditions have been included, as appropriate. Melting of lead shielding is generally not acceptable. Loss of external shielding, such as that sometimes used for neutron attenuation in spent-fuel packages, may be acceptable if it produces no other deleterious effects on the package and if the external radiation levels remain within allowable limits.

If the shielding model considers a homogenous source region (rather than a detailed heterogeneous model of the contents), ensure that such an approach is justified, and verify that the homogenized mass densities are correct. Atom densities should also be confirmed if used as input to shielding calculations.

5.3.4 Shielding Evaluation

The review of the shielding evaluation presented in the SARP should consider that §71.87(j) requires actual external radiation levels to be measured prior to shipment in order to verify that the limits of §71.47 are not exceeded. Other factors that should be considered in determining the level of effort for the shielding review include the expected magnitude of the radiation levels, the margin between calculations and regulatory limits, similarity with previously reviewed packages, thoroughness of the review of source terms and other input data, and bounding assumptions in the analysis.

5.3.4.1 Methods

Ensure that the methods used for the shielding evaluation are appropriate. Well-known computer programs should be referenced. Other codes or methods should be described in the SARP, and appropriate supplemental information should be provided. Verify that the number of dimensions of the code is appropriate for the package geometry, including streaming paths, if applicable.

Confirm that the cross-section library used by the code is applicable for the shielding calculations. Ensure that the code accounts for subcritical multiplication and secondary gamma production unless these conditions have been otherwise appropriately considered (e.g., in the source-term specification).

5.3.4.2 Input and Output Data

Verify that key input data for the shielding calculations are identified. These data will depend on the type of code (e.g., deterministic or Monte Carlo), as well as the code itself. The SARP should also include representative input files used in the analyses. Verify, as appropriate, that the information from the shielding models is properly input into the code.

At least one representative output file (or key sections of the file) should generally be included in the SARP. Ensure that proper convergence is achieved and that the calculated radiation levels in the output files agree with those reported in the text.

5.3.4.3 Flux-to-Dose-Rate Conversion

Ensure that the evaluation properly converts the gamma and neutron fluxes to dose rates. This conversion should generally use ANSI/ANS 6.1.1-1977 although other conversions may be used for point-kernel gamma calculations. Use of the conversions in ANSI/ANS 6.1.1-1991[2] can result in a significant underestimation of external radiation levels (as defined by 49 CFR 173.403 and 10 CFR 20.1004). In addition, the dose rates determined with the 1991 standard do not correspond physically to dose rates measured by typical radiation monitoring instruments.

Verify the accuracy of the flux-to-dose rate conversion factors, which should be tabulated as a function of the energy group structure used in the shielding calculation.

5.3.4.4 External Radiation Levels

Confirm that the external radiation levels under normal conditions of transport and hypothetical accident conditions agree with the summary tables discussed in Section 5.3.1.3 and that they meet the limits in §71.47(a) or §71.47(b), and §71.51(a)(2), as applicable. Verify that the analysis shows that the locations selected are those of maximum dose rates. To determine maximum dose rates, radiation levels may be averaged over the cross-sectional area of a probe of reasonable size.[3] For packages with streaming paths or voids, averaging should not be used to reduce the radiation levels resulting from such features. Averaging is also not acceptable for assessing cracks, pinholes, uncontrollable voids, or other defects as required by §71.85(a).

Ensure that the external radiation levels are reasonable and that their variations with location are consistent with the geometry and shielding characteristics of the package. Verify that the radiation levels presented in the shielding evaluation section are consistent with those in the summary table reviewed in Section 5.3.1.3 above.

Confirm that the evaluation addresses damage to the shielding under normal conditions of transport and hypothetical accident conditions. Verify that any damage under normal conditions of transport (§71.71) does not result in a significant increase in the external dose rates, as required by §71.43(f) and §71.51(a)(1). Any increase should be explained and justified as not significant.

5.3.5 Appendix

Confirm that the appendix includes a list of references, copies of applicable references if not generally available to the reviewer, computer code descriptions, input and output files, test results, flux-to-dose-rate conversion factors, and other appropriate supplemental information.

5.4 Evaluation Findings

5.4.1 Findings

The review should ensure that the information presented supports a conclusion that the regulatory requirements in Section 5.2 above are satisfied.

The TRR should include a finding similar to the following:

> Based on review of the statements and representations in the SARP, the staff concludes that the shielding design has been adequately described and evaluated and that the package meets the external radiation requirements of 10 CFR Part 71.

5.4.2 Conditions of Approval

The TRR should clearly identify any conditions of approval that should be included in Section 5 of the certificate of compliance. In addition to specifications of authorized contents and information specified on the engineering drawings, other conditions of approval applicable to the Shielding Evaluation chapter of the SARP may include:

- Restriction for exclusive-use shipment
- Limitations on vehicle dimensions or package position/orientation for exclusive-use shipments
- Requirement for personnel in normally occupied positions of the vehicle to wear dosimetry devices in accordance with 10 CFR 20.1502.

5.5 References

1. American Nuclear Society, "American National Standard for Neutron and Gamma-Ray Flux to Dose Rate Factors," ANSI/ANS 6.1.1-1977, LaGrange Park, Illinois.
2. American Nuclear Society, "American National Standard for Neutron and Gamma-Ray Fluence to Dose Factors," ANSI/ANS 6.1.1-1991, LaGrange Park, Illinois.
3. U.S. Nuclear Regulatory Commission, "Averaging of Radiation Levels Over the Detector Probe Area," HPPOS-13, in Health Physics Positions Data Base, NUREG/CR-5569, Rev. 1, 1992.

6 CRITICALITY REVIEW

This review verifies that the package design meets the criticality safety requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

The Criticality review is based in part on the descriptions and evaluations presented in the General Information, Structural Evaluation, and Thermal Evaluation chapters of the SARP. Similarly, the results of the Criticality review are considered in the review of the Operating Procedures, the Acceptance Tests and Maintenance Program, and Quality Assurance. An example of this information flow for the Criticality review is shown in Figure 6-1.

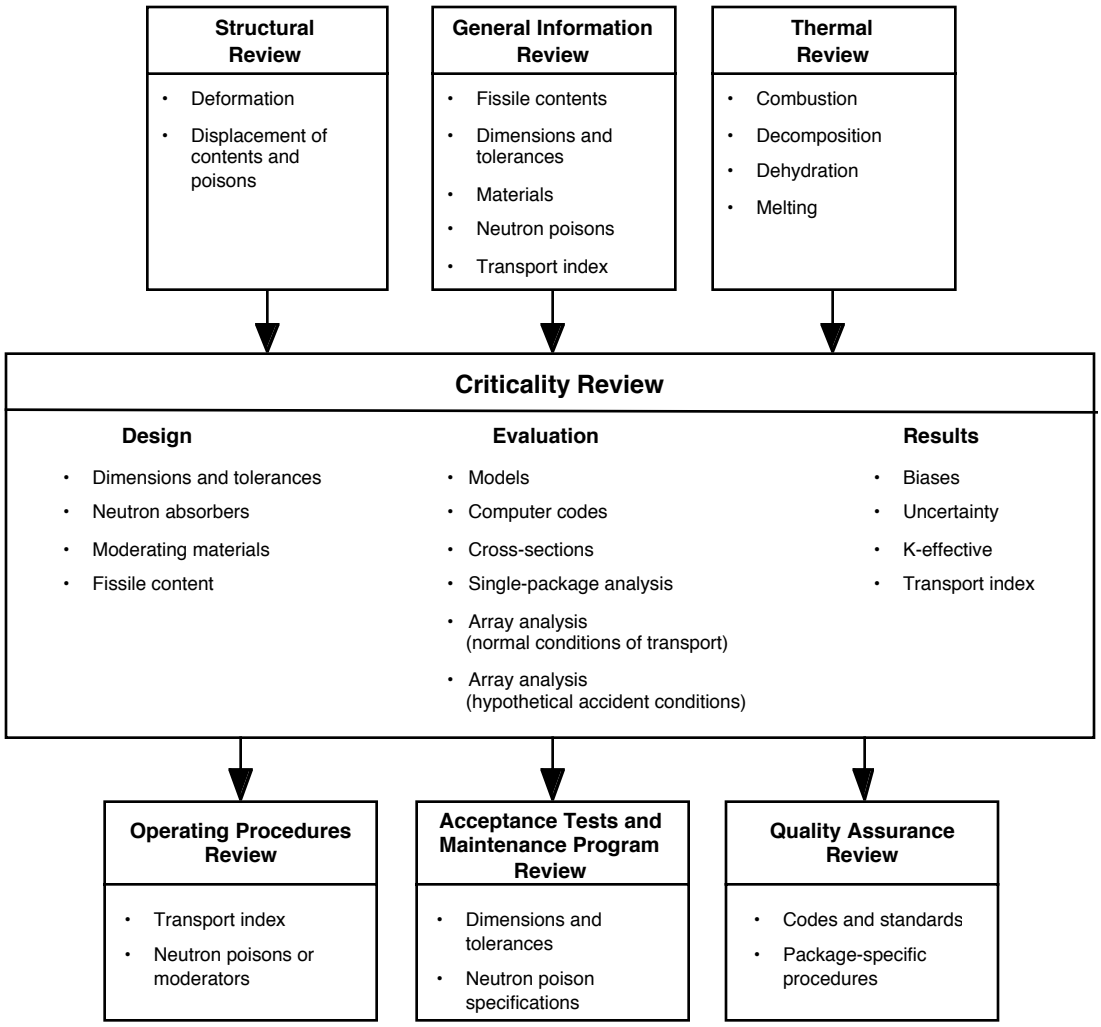

Figure 6-1 Example of Information Flow for the Criticality Review

6.1 Areas of Review

The description and evaluation of the criticality design should be reviewed. The criticality review should include the following:

6.1.1 Description of Criticality Design

- Design Features
- Codes and Standards
- Summary Table of Criticality Evaluations

6.1.2 Fissile Material and Other Contents

6.1.3 General Considerations for Criticality Evaluations

- Model Configuration
- Material Properties
- Demonstration of Maximum Reactivity
- Computer Codes and Cross-Section Libraries

6.1.4 Single Package Evaluation

- Configuration
- Results

6.1.5 Evaluation of Undamaged-Package Arrays (Normal Conditions of Transport)

- Configuration
- Results

6.1.6 Evaluation of Damaged-Package Arrays (Hypothetical Accident Conditions)

- Configuration
- Results

6.1.7 Transport Index for Nuclear Criticality Control

6.1.8 Benchmark Evaluations

- Applicability of Benchmark Experiments
- Bias Determination

6.1.9 Appendix

6.2 Regulatory Requirements

Regulatory requirements of 10 CFR Part 71 applicable to the Criticality review of fissile material packages are as follows:

- The package design must be described and evaluated to demonstrate that it meets the criticality requirements of 10 CFR Part 71. [§71.31(a)(1), §71.31(a)(2), §71.33, §71.35(a)]
- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]
- Unknown properties of fissile material must be assumed to be those which will credibly result in the highest neutron multiplication. [§71.83]
- A single package must be subcritical under the conditions of §71.55(b), §71.55(d), and §71.55(e).
- The package must be designed, constructed, and prepared for shipment so that there will be no significant reduction in the effectiveness of the packaging under the tests specified in §71.71 for normal conditions of transport. [§71.43(f), §71.51(a)(1), §71.55(d)(4)]
- An array of undamaged packages must be subcritical under the conditions of §71.59(a)(1).
- An array of damaged packages must be subcritical under the conditions of §71.59(a)(2).
- A fissile material package must be assigned a transport index for nuclear criticality control to limit the number of packages in a single shipment. [§71.59, §71.35(b)]

6.3 Review Procedures

The following procedures are generally applicable to the review of the Criticality Evaluation chapter of the SARP. These procedures correspond to the Areas of Review listed in Section 6.1 of this PRG.

6.3.1 Description of Criticality Design

6.3.1.1 Design Features

Review the General Information chapter of the SARP and any additional description of the criticality design presented in the Criticality Evaluation chapter. Design features important for criticality include:

- Dimensions and tolerances of the containment system for fissile material
- Structural components that maintain the fissile material or neutron poisons in a fixed position within the package or in a fixed position relative to each other

- Locations, dimensions, and densities (concentration) of neutron absorbing materials and moderating materials, including neutron poisons and shielding
- Dimensions and tolerances of floodable voids and flux traps within the package
- Dimensions and tolerances of the overall package that affect the physical separation of the fissile material contents in package arrays.

Confirm that the text and sketches describing the criticality design features are consistent with the engineering drawings and the models used in the criticality evaluation.

6.3.1.2 Codes and Standards

Verify that any codes or standards applicable to the criticality design of the package are identified and appropriate, including those for material specifications and fabrication. Ensure that such codes and standards are consistent with those specified in the General Information, Structural, and Thermal Evaluation chapters of the SARP. Determine if these codes or standards specify temperature limits for materials.

If codes, standards, or similar documents that provide subcritical limits are used in the criticality evaluation, ensure that the conditions specified in those documents are applicable to a package or array of packages under normal conditions of transport and hypothetical accident conditions.

6.3.1.3 Summary Table of Criticality Evaluation

Review the summary table of the criticality evaluation, which should address the following cases, as described in Sections 6.3.4 through 6.3.6:

- A single package, under the conditions of §71.55(b), §71.55(d), and §71.55(e)
- An array of undamaged packages, under the conditions of §71.59(a)(1)
- An array of damaged packages, under the conditions of §71.59(a)(2).

Verify that the table shows that the maximum multiplication factor for each case, including all uncertainties and the bias from benchmark calculations, does not exceed 0.95. (The administrative margin should be 0.05.) The table should include the number of packages evaluated and a brief description of the conditions of the package and array, as applicable. Because of the requirements of §71.43(f), the condition of an undamaged package should be that of a package subjected to the tests for normal conditions of transport. Table 6.1 illustrates an example table summarizing calculations performed with a Monte Carlo code. The terminology for the uncertainties and bias in Table 6.1 is consistent with that in NUREG/CR-5661[1] and NUREG/CR-6361.[2] Because variations in the details of bias determination have been used over the years, the reviewer should ensure that the approach is adequately described. See Section 6.3.8 of this PRG.

Review of the transport index for nuclear criticality control, as listed in the summary table, is discussed in Section 6.3.7 below.

Table 6.1 Example of Summary Table for Criticality Evaluations

Type of Evaluation/ Package Condition	No. of Packages*	$k + 2\sigma$ (package or array)	Bias (β)	Uncertainty in bias ($\Delta\beta$)	$k + 2\sigma - \beta$ ** $+\Delta\beta$
Single Package (Description of package condition)	1				
Undamaged Array (Description of package condition, array configuration)					
Damaged Array (Description of package condition, array configuration)					

* Transport Index for Nuclear Criticality Control = _____.

** Positive biases are not subtracted.

6.3.2 Fissile Material and Other Contents

Ensure that the specifications for the contents used in the criticality evaluation are consistent with those in the General Information chapter of the SARP. Specifications relevant to the criticality evaluation include fissile material mass, dimensions, enrichment or isotopic composition, physical and chemical form, density, moisture, and other characteristics depending on the specific contents. In addition, nonfissile materials used as moderators and absorbers must be specified if they are to be included as authorized contents in the certificate of compliance.

Specifications for fuel assemblies and rods should include:

- Type of fuel assemblies or rods and vendor/model, as appropriate
- Dimensions/tolerances of fuel (including annular pellets), cladding, fuel-cladding gap, pitch, and rod length
- Number of rods per assembly, and locations and dimensions of guide tubes and burnable poisons (see Section 6.3.3.2)
- Materials and densities
- Active fuel length
- Enrichment (variation by rod if applicable) before irradiation (see below)
- Chemical and physical form
- Mass of initial heavy metal per assembly or rod
- Number of fuel assemblies or individual rods per package
- Other information affecting the criticality evaluation, as applicable.

To date, burnup credit (to account for depletion of fissile material or increase in fission product poisons due to irradiation) has been accepted only on a very limited basis,[3] which is generally not applicable to material shipped by DOE. Consequently, the enrichment for spent fuel should be that of the unirradiated fuel, except in rare cases where irradiated material has a higher reactivity. If assemblies contain fuel with several enrichments, the evaluation should either assume the maximum enrichment or demonstrate that another approach (e.g., average enrichment) is bounding. Section 6.3.3.2 discusses consideration of poison densities and the depletion of burnable poisons.

Any differences in the contents specifications from those in the General Information chapter should be clearly identified and justified.

Because a partially filled container may allow more physical space for moderators (e.g., water), the most reactive case is not necessarily that with the maximum allowable contents. Fuel rods that have been removed from an assembly should be replaced with dummy rods that displace an equal amount of water unless the criticality analysis considers the additional moderation resulting from their absence. The requirement for dummy rods, if applicable, should be specified as a condition of approval in the certificate of compliance.

If the package is designed for multiple types of contents, the SARP may include a separate criticality evaluation and propose different criticality controls for each contents type. Any assumptions that certain contents need not be evaluated because they are less reactive than those evaluated should be properly justified.

6.3.3 General Considerations for Criticality Evaluations

The considerations discussed below are applicable to the review of criticality evaluations of a single package and arrays of packages under normal conditions of transport and hypothetical accident conditions.

General guidance for preparing criticality evaluations of transportation packages is provided in NUREG/CR-5661.

6.3.3.1 Model Configuration

Examine the Structural and Thermal Evaluation chapters of the SARP to determine the effects of the normal conditions of transport and hypothetical accident conditions on the packaging and its contents. Verify that the models used in the criticality evaluation are consistent with these effects and with the engineering drawings. Coordinate with the Structural and Thermal reviews as appropriate.

Review the configuration and dimensions of the contents and packaging used in the criticality models. For some types of packagings and contents (e.g., powders), the contents can be positioned at various locations and densities. The relative location and physical properties of the contents within the packaging should be justified as those that result in the maximum reactivity.

Ensure that the SARP considers deviations from nominal design configurations in the manner which maximizes reactivity. Examples of such deviations include:

- Dimensional tolerances, e.g., for cavity sizes and poison thickness
- Off-centered positioning of contents within the containment vessel or spent-fuel basket
- Off-centered positioning of basket or containment vessel within the package
- Preferential flooding of regions within the package.

Determine if the SARP includes any specifications regarding the condition of the contents. If the contents permit damaged fuel, the maximum extent of damage should be specified and addressed in the criticality analyses, as appropriate. Additional information on canning of damaged fuel is discussed in Section 4.3.1.3 of this PRG.

The contents of some packages (e.g., fuel assemblies) may be in the form of a finite lattice. With current computational capability, homogenization of the fissile region should generally be avoided. If a homogenized configuration is used, the SARP should demonstrate its appropriateness (e.g., by comparing k_{eff} of heterogeneous and homogeneous models and by consistently evaluating benchmark experiments).

6.3.3.2 Material Properties

Verify that the appropriate mass densities and atom densities are provided for materials used in the models of the packaging and contents. Material properties should be consistent with the condition of the package under the tests of §71.71 and §71.73, and any differences between normal conditions of transport and hypothetical accident conditions should be addressed.

Ensure that materials relevant to the criticality design (e.g., poisons, foams, plastics, and other hydrocarbons) are properly specified and the data sources referenced. Verify that materials will not degrade during the service life of the packaging. No more than 75% of the specified minimum neutron poison concentration in packaging components or in unirradiated contents should generally be considered in the criticality evaluation. No credit should be taken for burnable poisons in irradiated contents (e.g., spent fuel).

Unknown properties of fissile material must be assumed to be those which will credibly result in the highest neutron multiplication.

6.3.3.3 Demonstration of Maximum Reactivity

Verify that the analyses evaluate the most reactive configuration of each case listed in Section 6.3.1.3 (single package, array of undamaged packages, and array of damaged packages). Assumptions and approximations should be clearly identified and justified.

Ensure that the analysis determines the optimum combination of internal moderation (within the package) and interspersed moderation (between packages), as applicable. Confirm that preferential flooding of different regions within the package, including the fuel-cladding gap, is considered as appropriate. As noted in Section 6.3.2, the maximum allowable fissile material is not necessarily the most reactive contents.

Additional guidance on determining the most reactive configurations is presented in NUREG/CR-5661 and in Sections 6.3.4 to 6.3.6 below.

6.3.3.4 Computer Codes and Cross-Section Libraries

Confirm that an appropriate computer code (or other acceptable method) is used for the criticality evaluation. Well-known codes should be clearly referenced. Other codes or methods should be described in the SARP, and appropriate supplemental information should be provided.

Ensure that the criticality evaluations use an appropriate cross-section library. If multigroup cross sections are used, confirm that the neutron spectrum of the package has been appropriately considered and that the cross sections are properly processed to account for resonance absorption and self-shielding. Additional information regarding cross-sections is provided in NMSS Information Notice No. 91-26[4] and NUREG/CR-6328.[5]

Confirm that the computer code has been properly used in the criticality evaluation. Key input data for the criticality calculations should be identified. Depending on the code used, these data include number of neutrons per generation, number of generations, convergence criteria, mesh selection, etc. The SARP should include at least one representative input file for a single package, undamaged array, and damaged array evaluation. Verify, as appropriate, that the information from the criticality model, material properties, and cross sections is properly input into the code.

An output file (or key sections) should generally be included in the SARP for each representative input file. Ensure that the calculations have properly converged and that the calculated multiplication factors from the output files agree with those reported in the evaluation.

The review should generally include a detailed confirmatory analysis of the criticality calculations reported in the SARP. As a minimum, perform an independent calculation of the most reactive case, as well as sensitivity analyses to confirm that the most reactive case has been correctly identified. To the extent practical, use an independent model of the package and a different code and cross-section set from that of the SARP evaluation.

6.3.4 Single Package Evaluation

6.3.4.1 Configuration

Ensure that the criticality evaluation analyzes a single package under the most reactive condition of §71.55(d) (normal conditions of transport) and §71.55(e) (hypothetical accident conditions), with water moderation as required by §71.55(b). The evaluations should consider:

- Fissile material in its most reactive credible configuration consistent with the condition of the package and the chemical and physical form of the contents
- Water moderation to the most reactive credible extent, including water inleakage to the containment system
- Full water reflection on all sides of the package, including close reflection of the containment system or reflection by the package materials, whichever is more reactive.

Verify that the package also meets the specifications of §§71.55(d)(2) through 71.55(d)(4) under normal conditions of transport. Coordinate with the Structural review.

6.3.4.2 Results

Confirm that most reactive single-package conditions are evaluated and that the results are consistent with the information presented in the summary table discussed in Section 6.3.1.3. If the package is shown to be subcritical by reference to a standard such as ANSI/ANS 8.1[6] in lieu of calculations, verify that the standard is applicable to the package conditions.

6.3.5 Evaluation of Undamaged-Package Arrays (Normal Conditions of Transport)

6.3.5.1 Configuration

Ensure that the criticality evaluation analyzes an array of 5N undamaged packages. N may not be less than 0.5. The evaluation should consider:

- The most reactive configuration of the array (e.g., pitch, package orientation, and shape of the array) with nothing between the packages
- The most reactive credible configuration of the packaging and its contents under normal conditions of transport. If the evaluation of the water spray test has demonstrated that water would not leak into the package, water inleakage need not be assumed.
- Full water reflection on all sides of a finite array.

6.3.5.2 Results

Confirm that the most reactive array conditions are evaluated and that the results of the analysis are consistent with the information presented in the summary table discussed in Section 6.3.1.3.

6.3.6 Evaluation of Damaged-Package Arrays (Hypothetical Accident Conditions)

6.3.6.1 Configuration

Ensure that the criticality evaluation analyzes an array of 2N damaged packages. N may not be less than 0.5. The evaluation should consider:

- The most reactive configuration of the array (e.g., pitch, package orientation, internal moderation, and shape of the array)
- Optimum interspersed hydrogenous moderation
- Full water reflection on all sides of a finite array
- The most reactive credible configuration of the packaging and its contents under hypothetical accident conditions.

The analysis of arrays of damaged packages should generally assume water inleakage into the individual packages (including the containment vessel). Demonstrating that an array of leaking packages remains subcritical is more straightforward than designing and demonstrating that a package does not leak. The immersion test of §71.73(c)(5) is not required if water inleakage is assumed in the criticality analysis.

If the array analysis assumes that water does not leak into the packages in arrays, the SARP should clearly justify the basis for that assumption, and the package evaluation should adequately

demonstrate that the package can reliably exclude water when it is subjected to the hypothetical accident condition tests in §71.73. The justification for neglecting water inleakage should show, at a minimum, that:

1. No inleakage of water occurs when the package is subjected to the immersion tests of §§71.73(c)(5) and 71.73(c)(6).
2. The testing or analysis clearly demonstrates that the most unfavorable conditions for water inleakage have been addressed (e.g., initial test conditions, orientations for drop, crush, puncture, fire, and water immersion tests).
3. The package is designed and fabricated in accordance with accepted codes and standards.
4. If the package is evaluated by analysis, the design margin is in accordance with these codes and standards. If the package is evaluated by testing, the effects of the tests on the condition of the package can be consistently reproduced and demonstrate an adequate margin of safety.
5. The quality and characteristics of the tested package are representative of, and no better than, actual packages fabricated in accordance with the design specifications.
6. The design leakage rate for the package is sufficient to preclude water inleakage under both normal conditions of transport and hypothetical accident conditions.
7. The package is maintained and periodically inspected to ensure that its performance during its service life is representative of the package evaluated in the application. Fabrication, maintenance, and periodic leakage tests are conducted in accordance with ANSI N14.5.[7]
8. The package is tested prior to each shipment to show that the leakage rate is less than that which would allow inleakage of water.
9. The sensitivity of the criticality analysis to water inleakage is addressed as appropriate. For example, would water inleakage into most packages in a large array be required before criticality could be achieved, or would an array with only a few leaking packages be critical?
10. Any other issues relevant to reliably precluding water inleakage are addressed as appropriate.

6.3.6.2 Results

Confirm that the most reactive array conditions are evaluated and that the results of the analysis are consistent with the information presented in the summary table discussed in Section 6.3.1.3.

6.3.7 Transport Index for Nuclear Criticality Control

Based on the number of packages demonstrated to be subcritical in the array analyses reviewed in Sections 6.3.5 and 6.3.6, verify that the SARP has determined the appropriate value of N and has calculated the criticality transport index in accordance with §71.59. The appropriate N must be the smaller value which assures subcriticality for both 5N packages under normal conditions of transport and 2N packages under hypothetical accident conditions. Note that due to round-off and differences between exclusive and nonexclusive use, N is not necessarily the number of packages that can be included in a shipment.

Ensure that the transport index is consistent with that reported in the summary table of Section 6.3.3 above and in the General Information chapter of the SARP. This transport index is typically specified in the certificate of compliance as the minimum transport index.

6.3.8 Benchmark Evaluations

Ensure that the computer codes for criticality calculations are benchmarked against critical experiments. Verify that the analysis of the benchmark experiments uses the same computer code, computer hardware, and cross-section library as those used to calculate the k_{eff} values for the package.

Additional guidance on benchmarking of nuclear criticality codes is provided in NUREG/CR-6361. Numerous well documented benchmark experiments have been published by the Nuclear Energy Agency, Organization for Economic Co-Operation and Development.[8]

6.3.8.1 Applicability of Benchmark Experiments

Review the general description of the benchmark experiments and confirm that they are appropriately referenced.

Verify that the benchmark experiments are applicable to the actual packaging design and contents. The benchmark experiments should have, to the maximum extent possible, the same materials, neutron spectra, and configuration as the package evaluations. Key package parameters that should be compared with those of the benchmark experiments include type of fissile material, enrichment, moderator-to-fissile ratio, poison, and configuration. Confirm that differences between the package and benchmarks are identified and properly considered.

In addition, the SARP should address the overall quality of the benchmark experiments and the uncertainties in experimental data (e.g., mass, density, dimensions). Ensure that these uncertainties are treated in a conservative manner, i.e., they result in a lower multiplication factor for the benchmark experiment.

6.3.8.2 Bias Determination

Examine the results of the calculations for the benchmark experiments and the method used to account for biases, including the contribution from uncertainties in experimental data.

Ensure that a sufficient number of applicable benchmark experiments are analyzed and that the results of these benchmark calculations are used to determine an appropriate bias for the package calculations. Statistical and convergence uncertainties of both benchmark and package calculations should be addressed. Confirm that the benchmark evaluations address trends in the bias with respect to parameters such as moderator-to-fissile ratio, pitch-to-rod diameter, assembly separation, neutron absorber material, etc. As indicated in Table 6.1, positive biases should not be used to reduce the calculational uncertainty. Additional information on determining biases and their range of applicability is provided in NUREG/CR-5661 and NUREG/CR-6361.

6.3.9 Appendix

Confirm that the appendix includes a list of references, copies of applicable references if not generally available to the reviewer, computer code descriptions, input and output files, test results, and any other appropriate supplemental information.

6.4 Evaluation Findings

6.4.1 Findings

The review should ensure that the information presented supports a conclusion that the regulatory requirements in Section 6.2 above are satisfied.

The TRR should include a finding similar to the following:

> Based on review of the statements and representations in the SARP, the staff concludes that the nuclear criticality safety design has been adequately described and evaluated and that the package meets the nuclear criticality safety requirements of 10 CFR Part 71.

6.4.2 Conditions of Approval

The TRR should clearly identify any conditions of approval that should be included in Section 5 of the certificate of compliance. In addition to specifications of authorized contents and information specified on the engineering drawings, other conditions of approval applicable to the Criticality Evaluation of the SARP may include:

- Minimum transport index
- Restriction for exclusive-use shipment
- Requirement to have specific neutron absorbers in place
- Requirement to replace vacant positions in fuel assemblies with dummy rods
- Specification of the allowed extent of damage for spent fuel.

6.5 References

1. U.S. Nuclear Regulatory Commission, “Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages,” NUREG/CR-5661, April 1997.
2. U.S. Nuclear Regulatory Commission, “Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages,” NUREG/CR-6361, January 1997.
3. U.S. Nuclear Regulatory Commission, “Limited Burnup Credit,” ISG-8, Rev. 1, Spent Fuel Project Office, 1999.
4. U.S. Nuclear Regulatory Commission, “Potential Nonconservative Errors in the Working Format Hansen-Roach Cross-Section Set Provided with the KENO and SCALE Codes, ” NMSS Information Notice No. 91-26, April 15, 1991.

5. U.S. Nuclear Regulatory Commission, "Adequacy of the 123-Group Cross-Section Library for Criticality Analyses of Water-Moderated Uranium Systems," NUREG/CR-6328, August 1995.

6. American Nuclear Society, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors," ANSI/ANS 8.1-1983 (R1998), LaGrange Park, Illinois.

7. American National Standards Institute, ANSI N14.5-1997, "American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment," New York.

8. Organization for Economic Co-Operation and Development, "International Handbook of Evaluated Criticality Safety Benchmark Experiments," NEA/NSC/Doc(95)03, Nuclear Energy Agency, September 1999.

7 OPERATING PROCEDURES REVIEW

This review verifies that the operating controls and procedures for the package meet the requirements of 10 CFR Part 71 and are adequate to assure that the package will be operated in a manner consistent with its evaluation for approval.

The Operating Procedures chapter of the SARP should establish the minimum steps necessary to assure safe performance of the package under normal conditions of transport and hypothetical accident conditions. Detailed procedures, or procedures unrelated to the safe operation of the package, should not be included. Commitments specified in the Operating Procedures chapter of the SARP are typically included by reference into the certificate of compliance as conditions of package approval. Consequently, operating procedures cannot be site-specific.

The Operating Procedures review is based in part on the descriptions and evaluations presented in the General Information, Structural Evaluation, Thermal Evaluation, Containment, Shielding Evaluation, and Criticality Evaluation chapters of the SARP. Similarly, results of the Operating Procedures review are considered in the Acceptance Tests and Maintenance Program review and in the Quality Assurance review. An example of the information flow for the Operating Procedures review is shown in Figure 7-1.

Because the Operating Procedures chapter of the SARP addresses information relevant to other SARP chapters, it should be reviewed by all review team members.

7.1 Areas of Review

Controls and procedures which assure that the package will be operated in a manner consistent with its evaluation for approval should be reviewed. The operating procedures review should include the following:

7.1.1 Package Loading

- Preparation for Loading
- Loading of Contents
- Preparation for Transport

7.1.2 Package Unloading

- Receipt of Package from Carrier
- Removal of Contents

7.1.3 Other Procedures

7.1.4 Preparation of Empty Package for Transport

7.1.5 Appendix

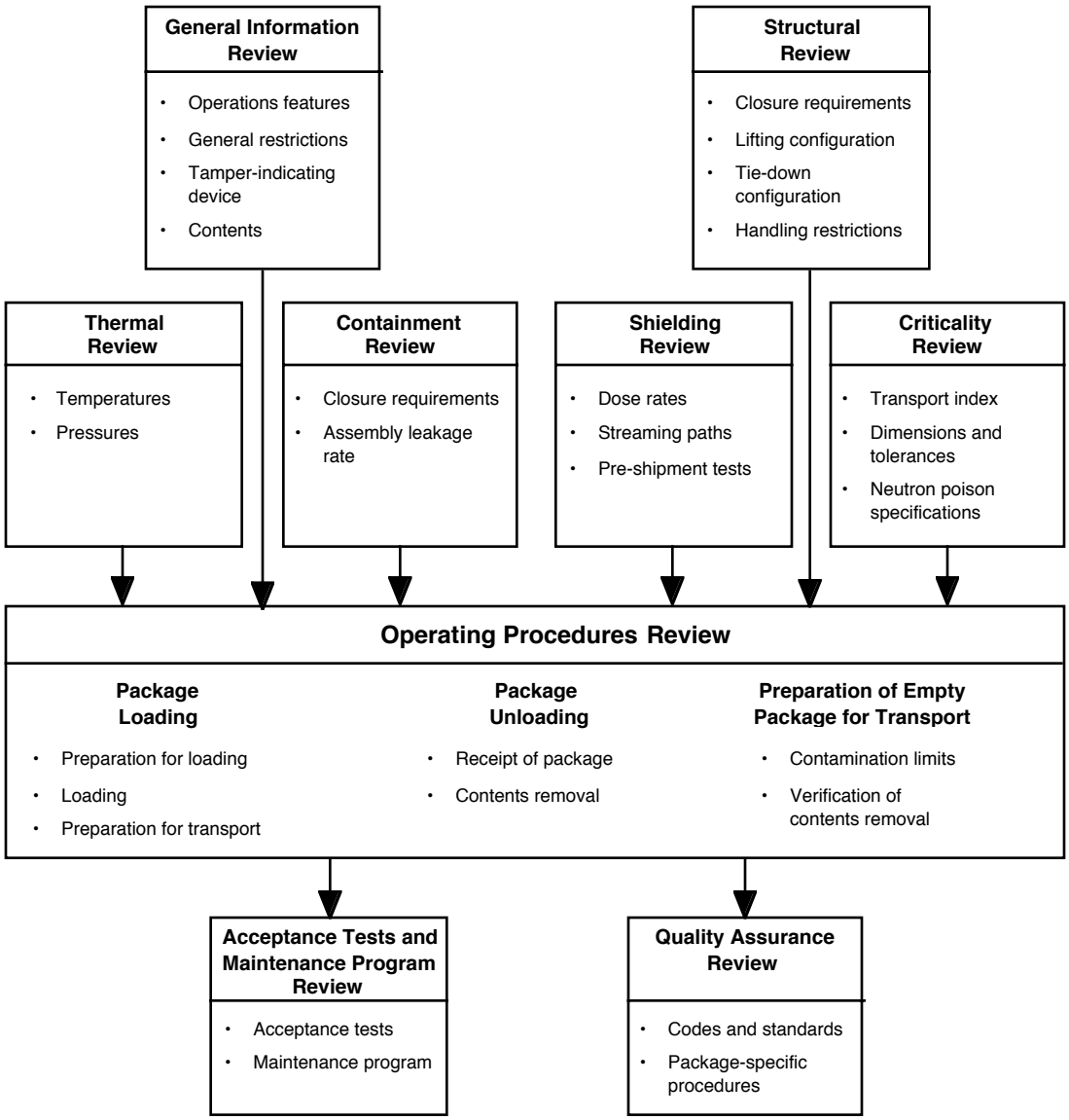

Figure 7-1 Example of Information Flow for the Operating Procedures Review

7.2 Regulatory Requirements

Regulatory requirements of 10 CFR Part 71 applicable to the Operating Procedures review are as follows:

- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]

- The application must include any special controls and precautions for transport, loading, unloading, and handling of a fissile material shipment, and any special controls in case of accident or delay. [§71.35(c)]
- A package must be conspicuously and durably marked with the model number, serial number, gross weight, and package identification number. [§71.85(c), §71.13(a), §71.13(b)]
- The application must include operating procedures which ensure that the package meets the routine-determination requirements of §71.87. [§71.81, 71.87]
- Unknown properties of fissile material must be assumed to be those which will credibly result in the highest neutron multiplication. [§71.83]
- Packages that require exclusive-use shipment because of increased radiation levels must be controlled by providing written instructions to the carrier. [§71.47(b-d)]
- The transport index of a package in a nonexclusive-use shipment must not exceed 10, and the sum of the transport indexes of all packages in the shipment must not exceed 50. [§71.47(a), §71.59(c)(1)]
- The sum of the transport indexes for nuclear criticality control of all packages in an exclusive-use shipment must not exceed 100. [§71.59(c)(2)]
- Prior to delivery of a package to a carrier, any special instructions needed to safely open the package must be provided to the consignee for the consignee's use in accordance with 10 CFR 20.1906(e). [§71.89]

7.3 Review Procedures

The following procedures are generally applicable to the review of the Operating Procedures chapter of the SARP. These procedures correspond to the Areas of Review listed in Section 7.1 of this PRG.

The operating procedures in the SARP should generally be listed in sequential order. Additional guidance on operating procedures is provided in NUREG/CR-4775.[1]

7.3.1 Package Loading

7.3.1.1 Preparation for Loading

Review the procedures for preparing the package for loading. At a minimum, the procedures should:

- Specify that the package should be loaded and closed in accordance with written procedures
- Describe any special controls and precautions for handling
- Verify that the package is in unimpaired physical condition and that all required periodic maintenance has been performed

- Ensure that the package is conspicuously and durably marked with the model number, serial number, gross weight, and package identification number
- Determine that the package is proper for the contents to be shipped, including the need for canning of damaged fuel or for a second containment vessel, if applicable
- Ensure that the use of the package complies with all other conditions of approval in the certificate of compliance.

7.3.1.2 Loading of Contents

Review the procedures for loading the contents. At a minimum, the procedures should:

- Identify any special handling equipment needed
- Describe any special controls and precautions for loading
- Indicate the method of loading the contents
- Ensure that any required moderator or neutron absorber is present and in proper condition
- Describe the method to remove water from the package, as appropriate
- Identify any requirement to vent gases from the package or add fill gas, as appropriate
- Ensure that each closure device of the package, including seals and gaskets, is properly installed, secured, and free of defects
- Verify that the bolt torques described in the procedures are consistent with those shown on the drawings
- Confirm that the package has been loaded and closed appropriately.

7.3.1.3 Preparation for Transport

Review the procedures for preparing the package for transport. At a minimum, the procedures should:

- Ensure that non-fixed (removable) radioactive contamination on external surfaces is as low as reasonably achievable, and within the limits specified in 49 CFR 173.443
- Describe the radiation survey to confirm that the allowable external radiation levels specified in §71.47 are not exceeded
- Describe the temperature survey to verify that limits specified in §71.43(g) are not exceeded
- Specify the assembly verification leakage rate, and ensure package closures are leak tested in accordance with ANSI N14.5[2]
- Ensure that any system for containing liquid is properly sealed and has adequate space or other specified provision for expansion of the liquid
- Verify that any pressure relief device is operable and set

- Ensure that any structural component that could be used for lifting or tie-down during transport is rendered inoperable for those purposes unless it meets the design requirements of §71.45
- Ensure that the tamper-indicating device is installed
- Specify the attachment of impact limiters, personnel barriers, or similar devices as applicable
- Describe, for a fissile material shipment, any special controls and precautions for transport, loading, unloading, and handling and any appropriate actions in case of an accident or delay which should be provided to the carrier or consignee
- Identify any special controls which should be provided to the carrier for a package shipped by exclusive use under the provisions of §71.47(b)(1)
- Identify any special controls which should be provided to the carrier for a fissile-material package in accordance with §71.35(c)
- Describe any special instructions which should be provided to the consignee for opening the package
- Ensure that the transport index for each package and the sum of the transport indexes for the shipment are appropriate for the type of shipment as appropriate.

7.3.2 Package Unloading

7.3.2.1 Receipt of Package from Carrier

Review the procedures for receiving the package. At a minimum, the procedures should:

- Ensure that the package is examined for visible damage, status of the tamper-indicating device, surface contamination, and external radiation levels
- Describe any special actions to be taken if the package is damaged, if the tamper-indicating device is not intact, or if surface contamination or radiation survey levels are too high
- Identify any special handling equipment needed
- Describe any proposed special controls and precautions for handling and unloading.

7.3.2.2 Removal of Contents

Review the procedures for removing the contents. At a minimum, the procedures should:

- Describe the appropriate method to open the package
- Identify the appropriate method to remove the contents
- Ensure that the contents are completely removed.

7.3.3 Other Procedures

Confirm that the SARP identifies other operational controls, as applicable. For example, some packages have a maximum allowable shipping duration due to potential generation of hydrogen gas.

7.3.4 Preparation of Empty Package for Transport

Review the procedures for preparing an empty package for transport. At a minimum, the procedures should:

- Verify that the package is empty
- Ensure that external and internal contamination levels meet the requirements of 49 CFR 173.443 and 49 CFR 173.428
- Describe the package closure requirements
- Identify any other special controls or procedures as appropriate.

7.3.5 Appendix

Confirm that the appendix includes a list of references, copies of applicable references if not generally available to the reviewer, test results, and other appropriate supplemental information.

7.4 Evaluation Findings

7.4.1 Findings

The review should ensure that the information presented supports a conclusion that the regulatory requirements in Section 7.2 above are satisfied.

The TRR should include a finding similar to the following:

> Based on review of the statements and representations in the SARP, the staff concludes that the operating controls and procedures for the package meet the requirements of 10 CFR Part 71 and are adequate to assure that the package will be operated in a manner consistent with its evaluation for approval.

7.4.2 Conditions of Approval

The TRR should clearly identify any conditions of approval that should be included in Section 5 of the certificate of compliance. The entire Operating Procedures chapter of the SARP is typically included by reference into the certificate of compliance as a condition of the package approval.

7.5 References

1. U.S. Nuclear Regulatory Commission, "Guide for Preparing Operating Procedures for Shipping Packages," NUREG/CR-4775, July 1988.
2. Institute of Nuclear Materials Management, "American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment," ANSI N14.5-1997, New York.

8 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM REVIEW

This review verifies that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71 and that the maintenance program is adequate to assure packaging performance during its service life.

The Acceptance Tests and Maintenance Program chapter of the SARP should establish the minimum steps necessary to assure that the package will perform throughout its service life in the manner in which it was evaluated. Detailed procedures or site-specific requirements should not be included. Commitments specified in the Acceptance Tests and Maintenance Program chapter of the SARP are typically included in the certificate of compliance as conditions of package approval.

The Acceptance Tests and Maintenance Program review is based in part on the descriptions and evaluations presented in previous chapters of the SARP. Similarly, the results of this review are considered in the Quality Assurance review. In addition, the review of other chapters of the SARP may depend on the Acceptance Test and Maintenance Program review (e.g., operating procedures for leakage testing prior to shipment may depend on the maintenance leakage test). An example of the information flow for this review is shown in Figure 8-1.

Because the Acceptance Tests and Maintenance Program chapter of the SARP addresses information relevant to other SARP chapters, it should be reviewed by all review team members.

8.1 Areas of Review

The description of the acceptance tests and maintenance program should be reviewed. The review should include:

8.1.1 Acceptance Tests

- Visual Inspections and Measurements
- Weld Examinations
- Component Tests
- Material Tests
- Structural and Pressure Tests
- Leakage Rate Tests
- Shielding Tests
- Thermal Tests
- Other Tests

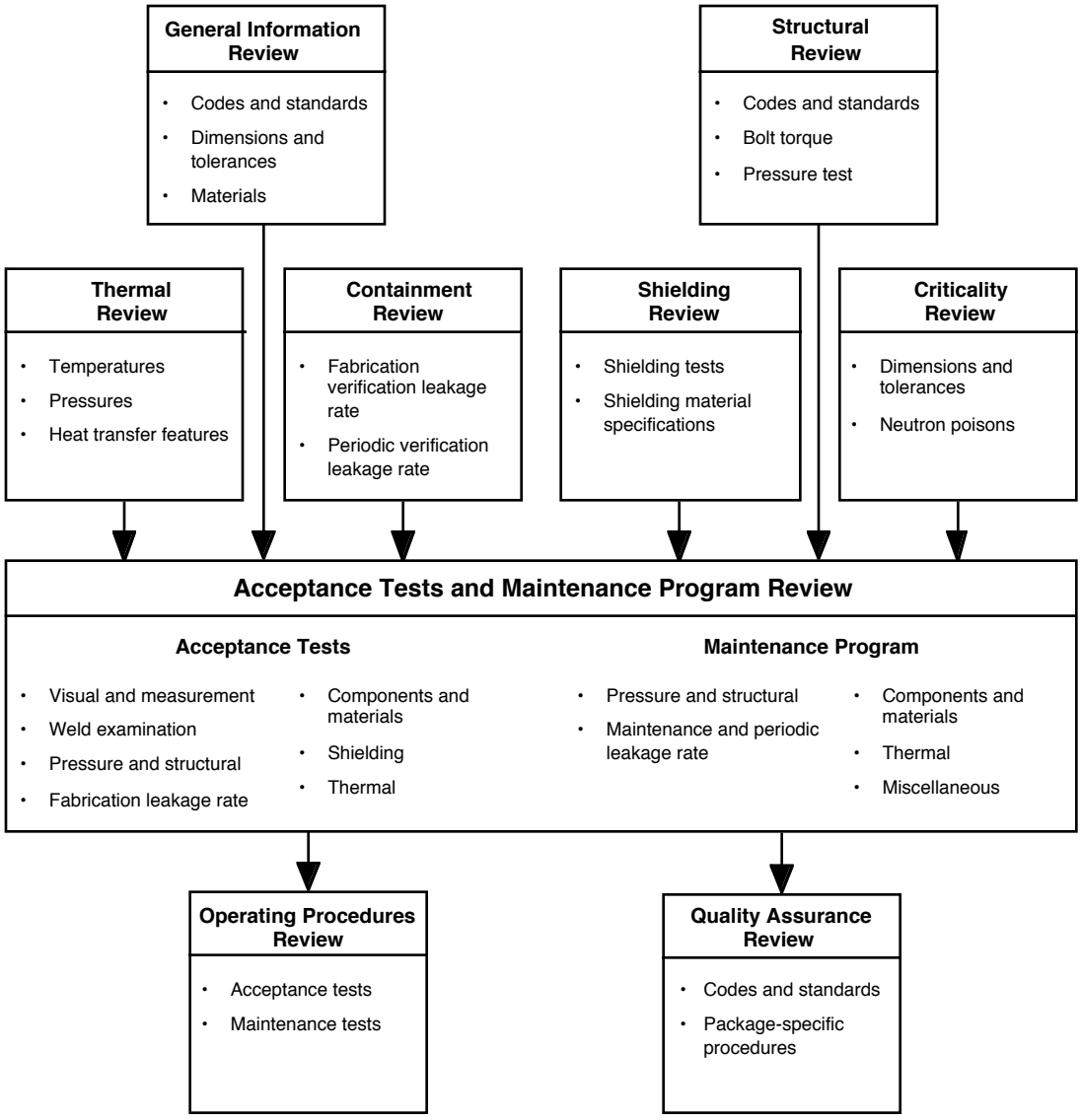

Figure 8-1 Example of Information Flow for the Acceptance Tests and Maintenance Program Review

8.1.2 Maintenance Program

- Component Tests
- Material Tests
- Structural and Pressure Tests
- Leakage Rate Tests
- Thermal Tests
- Other Tests

8.1.3 Appendix

8.2 Regulatory Requirements

Regulatory requirements of 10 CFR Part 71 applicable to the Acceptance Tests and Maintenance Program review are as follows:

8.2.1 Acceptance Tests

- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]
- Before first use, the fabrication of each packaging must be verified to be in accordance with the approved design. [§71.85(c)]
- Before first use, each packaging must be inspected for cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce its effectiveness. [§71.85(a)]
- Before first use, if the maximum normal operating pressure of a package exceeds 35 kPa (5 psi) gauge, the containment system of each packaging must be tested at an internal pressure at least 50% higher than maximum normal operating pressure to verify its ability to maintain structural integrity at that pressure. [§71.85(b)]
- Before first use, each packaging must be conspicuously and durably marked with its model number, serial number, gross weight, and a package identification number. [§71.85(c)]
- The licensee must perform any tests deemed appropriate by the certifying authority. [§71.93(b)]

8.2.2 Maintenance Program

- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]
- The packaging must be maintained in unimpaired physical condition except for superficial defects such as marks or dents. [§71.87(b)]
- The presence of any moderator or neutron absorber, if required, in a fissile material package must be verified prior to each shipment. [§71.87(g)]
- The licensee must perform any tests deemed appropriate by the certifying authority. [§71.93(b)]

8.3 Review Procedures

The following procedures are generally applicable to the review of the Acceptance Tests and Maintenance Program chapter of the SARP. These procedures correspond to the Areas of Review listed in Section 8.1 of this PRG.

8.3.1 Acceptance Tests

Verify that the following tests, as applicable, are to be performed prior to the first use of each package. Information presented on each test should include a description of the test and its acceptance criteria as appropriate. Applicable sections of the quality assurance program and procedures may be referenced if applicable.

Each package must be fabricated in accordance with the engineering drawings listed in the certificate of compliance.

Additional guidance on acceptance tests is provided in NUREG/CR-3854.[1]

8.3.1.1 Visual Inspections and Measurement

Ensure that inspections are performed to verify that the packaging has been fabricated and assembled in accordance with the engineering drawings. Dimensions and tolerances specified on the drawings should be confirmed by measurement.

8.3.1.2 Weld Examinations

Verify that welding examinations are performed to verify fabrication in accordance with the drawings, codes, and standards specified in the SARP. Location, type, and size of the welds should be confirmed by measurement. Other specifications for weld performance, nondestructive examination, and acceptance should be verified as appropriate. Additional guidance on welding criteria is provided in NUREG/CR-3019.[2]

8.3.1.3 Component Tests

Confirm that appropriate tests and acceptance criteria are specified for components that affect package performance. Examples of such components include seals, gaskets, valves, fluid transport systems, and rupture disks or other pressure-relief devices. Components should be tested to meet the performance specifications shown on the engineering drawing of the package. When tests adversely affect the continued performance of a component (e.g., rupture disks), applicable quality assurance procedures should be described to justify that the tested component is equivalent to the component that will be used in the packaging.

8.3.1.4 Material Tests

Verify that methods are in place to demonstrate that the materials meet the specifications shown on the engineering drawing of the package. Ensure that material examinations are performed in accordance with the codes and standards specified. Confirm that appropriate tests and acceptance criteria are specified for non-code materials. Tests for neutron absorbers (e.g., boron, gadolinia) and insulating materials (e.g., foams, fiberboard) should assure that minimum specifications for density and composition are achieved.

8.3.1.5 Structural and Pressure Tests

Verify that the structural or pressure tests are identified and described. Such tests should comply with §71.85(b), as well as applicable codes or standards specified in the SARP (e.g., in the Structural Evaluation chapter).

8.3.1.6 Leakage Rate Tests

Verify that the containment system of the packaging will be subjected to the fabrication leakage test specified in ANSI N14.5.[3] Verify that all closures, including drains and vents, are leak-tested. The acceptable leakage criterion should be consistent with that identified in the Containment chapter of the SARP.

8.3.1.7 Shielding Tests

Ensure that appropriate shielding tests are specified for both neutron and gamma radiation. The tests and acceptance criteria should be sufficient to assure that no voids or streaming paths exist in the shielding.

8.3.1.8 Thermal Tests

Verify that appropriate tests are specified to demonstrate the heat transfer capability of the packaging. These tests should confirm that the heat transfer performance determined in the evaluation is achieved in the fabrication process.

8.3.1.9 Other Tests

Verify that any additional tests are described, as applicable, to demonstrate that the package has been fabricated in accordance with its approved design. Confirm that tests specified in the SARP are sufficient to meet the requirements of §71.85(a). Verify that after the acceptance tests are completed, the package will be durably marked in accordance with §71.85(c).

8.3.2 Maintenance Program

Confirm that the maintenance program is adequate to assure that packaging effectiveness is maintained throughout its service life. Maintenance tests and inspections should be described with schedules for each test or replacement of parts and criteria for minor refurbishment and replacement of parts, as applicable.

8.3.2.1 Component Tests

Verify that periodic tests and replacement schedules for components are described, as appropriate. Elastomeric seals should generally be replaced and leak tested within the 12-month period prior to shipment. Metallic seals are generally replaced prior to each shipment.

8.3.2.2 Material Tests

Confirm that the SARP identifies any process that could result in deterioration of packaging materials, including loss of neutron absorbers, reduction in hydrogen content of shields, and density changes of insulating materials. Appropriate tests and their acceptance criteria to ensure packaging effectiveness for each shipment should be specified.

8.3.2.3 Structural and Pressure Tests

Verify that any periodic structural or pressure tests are identified and described. Such tests would generally be applicable to codes, standards, or other procedures specified in the SARP.

8.3.2.4 Leakage Rate Tests

Confirm that the containment system of the packaging will be subjected to the periodic and maintenance leakage rate tests specified in ANSI N14.5. The acceptable leakage rate criterion should be consistent with that identified in the Containment chapter of the SARP. Ensure that replacement schedules for seals are described, as appropriate.

8.3.2.5 Thermal Tests

Verify that periodic tests to assure the heat transfer capability during the service life of the packaging are described. Tests similar to the acceptance tests discussed in Section 8.3.1.8 may be applicable. The typical interval for periodic thermal tests is five years.

8.3.2.6 Other Tests

Confirm that any additional tests are described, as applicable, to demonstrate that the package will perform throughout its service life in accordance with its approved design.

8.3.3 Appendix

Confirm that the appendix includes a list of references, copies of applicable references if not generally available to the reviewer, and other appropriate supplemental information.

8.4 Evaluation Findings

8.4.1 Findings

The TRR should include a finding similar to the following:

> Based on review of the statements and representations in the SARP, the staff concludes that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71 and that the maintenance program is adequate to assure packaging performance during its service life.

8.4.2 Conditions of Approval

The TRR should clearly identify any conditions of approval that should be included in Section 5 of the certificate of compliance. The entire Acceptance Tests and Maintenance Program chapter of the SARP is typically included by reference into the certificate of compliance as a condition of package approval.

8.5 References

1. U.S. Nuclear Regulatory Commission, “Fabrication Criteria for Shipping Containers,” NUREG/CR-3854, March 1985.
2. U.S. Nuclear Regulatory Commission, “Welding Criteria for Use in the Fabrication of Radioactive Material Shipping Containers,” NUREG/CR-3019, March 1984.
3. Institute for Nuclear Materials Management, “American National Standard for Radioactive Material-Leakage Tests on Packages for Shipment,” ANSI N14.5-1997, New York.

9 QUALITY ASSURANCE REVIEW

This review verifies that the applicant has a quality assurance (QA) program that meets the requirements of 10 CFR Part 71 and that specific QA requirements for the package are adequate to assure that it is designed, fabricated, assembled, tested, used, maintained, modified, and repaired in a manner consistent with its evaluation in the SARP.

The QA chapter of the SARP should assure that adequate control is provided over all activities important to safety. The review focuses on two specific areas: (1) the applicant's QA program and (2) package-specific QA requirements applicable to all organizations that perform activities with the proposed package. Because the applicant's QA program description presented in the SARP is site-specific, it cannot be referenced in the certificate of compliance as a condition of approval. Package-specific QA requirements, however, are appropriate for all organizations and should be included as conditions of approval in the certificate of compliance. Note that Section 4 of the certificate specifies that package approval is also conditional on the fulfillment of the applicable QA requirements of 49 CFR Parts 100-199 and 10 CFR Part 71.

In addition to the QA-program requirements in Subpart H (Quality Assurance), 10 CFR Part 71 includes other quality-related provisions in Subpart D (Application for Package Approval), Subpart E (Package Approval Standards), Subpart F (Package, Special Form, and LSA-III Tests), and Subpart G (Operating Procedures). Consequently, other SARP chapters also address quality-related requirements, many of which are incorporated as conditions of approval in the certificate of compliance. For example, the drawings in the General Information chapter include dimensions and tolerances and codes or standards for fabrication and material specifications, and the requirements for operation, acceptance testing/maintenance are specified in the Operating Procedures chapter and in the Acceptance Tests and Maintenance Program chapter, respectively. The Structural, Thermal, Containment, Shielding, and Criticality Evaluation chapters may specify codes, standards, or other QA-related requirements that affect the package design, and the evaluation of the package design in these chapters addresses those components of the packaging that are important to safety. An example of the information flow for the QA review is shown in Figure 9-1.

Because the QA chapter of the SARP addresses information relevant to other SARP chapters, it should be reviewed by all review team members.

9.1 Areas of Review

The applicant's QA-program description and package-specific QA requirements should be reviewed. The QA review should include the following:

9.1.1 Description of Applicant's QA Program

- Scope
- Program Documentation and Approval
- Summary of 18 Quality Criteria
- Cross-Referencing Matrix

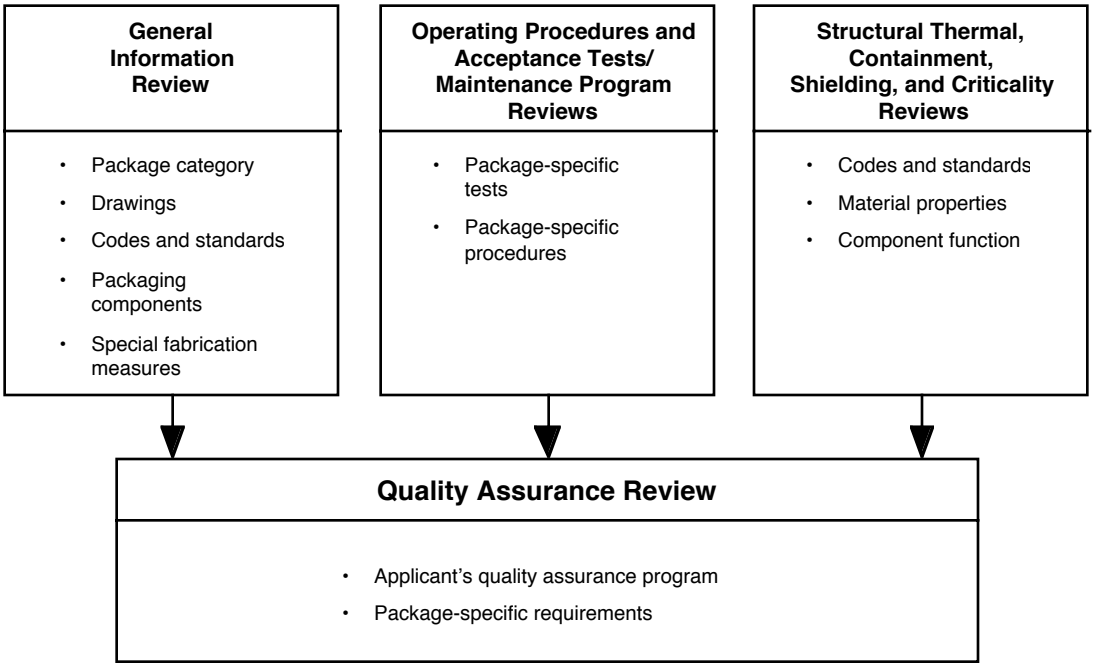

Figure 9-1 Example of Information Flow for the Quality Assurance Review

9.1.2 Package-Specific QA Requirements

- Graded Approach for Structures, Systems, and Components Important to Safety
- Package-Specific Quality Criteria and Package Activities

9.1.3 Appendix

9.2 Regulatory Requirements

Regulatory requirements of 10 CFR Part 71 applicable to the QA review are as follows:

- The application must describe the quality assurance program for the design, fabrication, assembly, testing, maintenance, repair, modification, and use of the package. [§71.31(a)(3), §71.37]
- The application must identify established codes and standards proposed for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of any codes and standards, the application must describe the basis and rationale used to formulate the package quality assurance program. [§71.31(c)]
- Package activities must be in compliance with the quality assurance requirements of Subpart H (§71.101-§71.137). A graded approach is acceptable. [§71.81, §71.101(b)]
- Sufficient written records must be maintained to furnish evidence of the quality of the packaging. These records include results of the determinations required by §71.85;

design, fabrication, and assembly records; results of reviews, inspections, tests, and audits; results of maintenance, modification, and repair activities; and other information identified in §71.91(c). Records must be retained for three years after the life of the packaging. [§71.91(c)]

- Records identified in §71.91(a) must be retained for three years after shipment of radioactive material. [§71.91(a)]
- Records must be available for inspection. Records are valid only if stamped, initialed, or signed and dated by authorized personnel or otherwise authenticated. [§71.91(b)]
- Any significant reduction in the effectiveness of a packaging during use must be reported to the certifying authority. [§71.95(a)]
- Details of any defects with safety significance in a package after first use, with the means employed to repair the defects and prevent their reoccurrence, must be reported. [§71.95(b)]
- Instances in which a shipment does not comply with the conditions of approval in the certificate of compliance must be reported to the certifying authority. [§71.95(c)]

9.3 Review Procedures

The following procedures are generally applicable to the review of the QA chapter of the SARP. These procedures correspond to the Areas of Review listed in Section 9.1 of this PRG.

9.3.1 Description of Applicant's QA Program

9.3.1.1 Scope

Confirm that the SARP identifies those package activities for which the applicant has QA-responsibility. These activities may include design, fabrication, assembly, acceptance testing, procurement, use, periodic inspection, maintenance, and repair. Applicants should be considered responsible if they perform, contract, or otherwise oversee the activity. Although applicants are typically responsible for packaging design, responsibility for other activities may be assigned to other DOE organizations. For example, the applicant may design, fabricate, assemble, and perform acceptance testing of a packaging, but another DOE organization may assume responsibility for its use, periodic inspection, and maintenance.

9.3.1.2 Program Documentation and Approval

Verify that the applicant has an approved QA program applicable to packaging. This will likely be an "umbrella" program that provides QA requirements for all quality-related packaging activities (i.e., not specific to the package submitted for approval). This program will also likely supplement the applicant's overall site QA program. The SARP should specify QA-program documentation by title, number, revision, and date. The approving organization, document, and date of approval should also be identified.

Confirm that the SARP specifies on which QA-requirements document (e.g., DOE O 414.1,[1] Subpart H of 10 CFR Part 71) the QA program and its approval are based. Although DOE

organizations are generally required to comply with DOE O 414.1* and 10 CFR 830.120, QA programs for packages must also comply with Subpart H (and other applicable subparts) of 10 CFR Part 71. The SARP should explicitly state that the QA program complies with Subpart H. Justification for this compliance, if not cited in the approval documentation, should be presented as discussed below. In general, QA program for packages approved under ASME NQA-1[2] or Appendix B, 10 CFR Part 50, will meet the requirements of Subpart H.

In addition to his umbrella QA program, the applicant will generally need to develop detailed QA procedures specific to the package proposed in the SARP. Depending on the applicant's scope of responsibility, these procedures might address design testing, implementation of material and fabrication requirements, control of vendor activities, acceptance tests, maintenance and operational requirements, and record keeping. The SARP should describe existing package-specific procedures and documentation and identify those that are intended to be prepared in the future. As a minimum, detailed procedures for all activities performed during SARP preparation should be completed as described in Regulatory Guide (RG) 7.10, Annex 1.

9.3.1.3 Summary of 18 Quality Criteria

The level of detail reviewed in this section depends on the type of approval applicable to the applicant's QA program. For example, if the applicant has a QA program that has been approved as meeting the requirements of Subpart H by DOE headquarters or the DOE field/operations office, significantly less review will be necessary than if the program is approved only in accordance with DOE O 414.1 or 10 CFR 830.120.

Verify that the SARP demonstrates compliance with each of the 18 criteria of Subpart H (§71.103 to §71.137) appropriate to the scope of the applicant's responsibilities, as reviewed in Section 9.3.1.1 above. Guidance on evaluating these criteria is provided in RG 7.10.[3] Annex 1 of RG 7.10 addresses quality assurance programs applicable to design, fabrication, assembly, and testing of packagings. Annex 2 addresses procurement, use, maintenance, and repair (except for radiographic exposure devices, which are discussed in Annex 3). Unless the applicant justifies otherwise, the QA review should be based on Annex 1 or Annex 2, as appropriate.

If the applicant's QA program for packaging augments a site program based on DOE O 414.1 or 10 CFR 830.120, the SARP should demonstrate compliance with the 18 criteria of Subpart H. The review should specifically address compliance with the requirement for audits (§71.137).

9.3.1.4 Cross-Referencing Matrix

Confirm that the SARP provides a cross-referencing index which demonstrates that each of the 18 criteria are addressed by written procedures. An example of such a matrix is presented in Table 1, RG 7.10. Because of the inter-relationship of the 18 criteria in Subpart H, more than one quality procedure will generally be applicable to each criterion.

Since information presented on the applicant's QA program is both site-specific and subject to modification, it cannot be incorporated directly as a condition of package approval in the certificate of compliance. Site-specific methods of accomplishing tasks and implementing quality cannot generally be imposed on other organizations involved with the packaging.

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* DOE O 414.1 replaces DOE 5700.6C, which may still be applicable because of contractual relationships.

Similarly, a revision to the site QA program, an organizational change, or renumbering of the program documentation should not necessitate a revision of the SARP. The requirement for the applicant to maintain an appropriate QA program is specified in Section 4 of the certificate.

9.3.2 Package-Specific QA Requirements

The SARP should describe QA requirements for the proposed package. Requirements should be based on a graded approach, considering the importance to safety of package structures, systems, components, and activities. The review should address controls necessary for design, fabrication, testing, operations, maintenance, and repair to assure that the package will meet the requirements of 10 CFR Part 71 during its service life. Importance to safety should be based primarily on the ability of the package to provide:

- Containment of radioactive material
- Subcriticality of fissile material
- Shielding of radiation.

The graded approach should consider the complexity and proposed use of the package and its components as described in §71.105(c).

9.3.2.1 Graded Approach for Structures, Systems, and Components Important to Safety

Verify that the SARP provides a package-specific listing (Q-List) of all structures, systems, and components (SSCs) important to safety and that these SSCs are consistent with the parts list or similar information presented in the packaging drawings. Justification should be provided for any item identified on the drawings but not defined as important to safety in the Q-list.

Confirm that the SARP identifies a quality category (e.g., A, B, C) for each SSC important to safety and that these categories are appropriately defined. Ensure that the assigned categories are properly justified based on their definition, the package type, and the safety function of each SSC. Coordinate with the review of other SARP chapters as appropriate. Appendix A of RG 7.10 provides guidance on defining quality categories and QA requirements. Definitions of typical categories and representative safety classifications for SSCs of transportation packagings are also presented in Table 2 and Table 5, respectively, of NUREG/CR-6407.[4]

9.3.2.2 Package-Specific Quality Criteria and Package Activities

Verify that the SARP addresses each of the 18 quality criteria in Subpart H as they apply to the proposed package. The SARP should identify for each criterion, as applicable, the appropriate level of effort for package activities based on their importance to safety. Guidance on QA requirements applicable to each category is provided in Appendix A of RG 7.10. Other guidance is presented in Chapter 4 of NUREG/CR-6407, which also describes typical design and fabrication records maintained for each QA category. Table 9.1 below identifies typical levels of effort for each of the 18 criteria of Subpart H that should be considered in the review, based on quality category. Note that the omission of Category C items from QA effort may not be appropriate if they involve a condition of approval specified in the certificate of compliance.

Table 9.1 Typical Level of QA Effort by Quality Category

QA Element/Level of Effort	Category A	Category B	Category C
1. QA Organization			
Responsibility established	X	X	X
Authority and duties written	X	X	X
QA functions executed	X	X	X
Reporting levels clearly defined	X	X	X
Independence from cost and schedule assured	X	X	X
2. QA Program			
Procedures written	X	X	X
Activities affecting quality controlled	X	X	X
Graded approach established	X	X	X
Indoctrination and training provided	X	X	X
3. Design Control			
Most stringent codes and standards	X		
Codes and standards	X	X	
Prototype test and/or analysis	X	X	
Formal design review	X	X	
Internal peer review	X	X	
Off-the-shelf items			X
Conditions of approval controlled	X	X	X
4. Procurement Document Control			
Traceabililty	X	X	
Qualified vendor lists	X		
Off-the-shelf items			X
5. Instructions, Procedures, and Drawings			
Written and documented	X	X	
Qualitative or quantitative acceptance criteria	X	X	
Changes to conditions of approval listed in certificate controlled	X	X	X
6. Document Control			
Controlled issue	X	X	
Controlled changes	X	X	

Table 9.1 Typical Level of QA Effort by Quality Category (cont.)

QA Element/Level of Effort	Category A	Category B	Category C
7. Control of Purchased Material, Equipment, and Services			
Source evaluation and selection	X		
Inspection at contractor	X		
Formal receiving inspection	X	X	
Audits or surveillance at vendor plants	X		
Evidence of QA at contractor	X	X	
Objective proof that all specifications are met	X	X	X
Incoming inspection for damage only			X
8. Identification and Control of Materials, Parts, and Components			
Positive identification and traceability	X		
Identification and traceability to heats, lots, or other groupings		X	
Identification to end use drawings			X
9. Control of Special Processes			
Welding, heat treating, and NDE performed by qualified personnel	X		
Qualification records and training of personnel	X		
Only specified critical operations by qualified personnel		X	
No special processes			X
10. Internal Inspection			
Documented inspection of all specifications	X		
Process monitoring if required by quality	X		
Examination, measurement, or test of material or processed product to assure quality	X	X	
Inspectors independent of those performing operations	X	X	
Qualified inspectors only	X	X	
Visual receiving inspection only			X

Table 9.1 Typical Level of QA Effort by Quality Category (cont.)

QA Element/Level of Effort	Category A	Category B	Category C
11. Test Control			
Written test program	X	X	
Written test procedures	X	X	
Documentation of testing and evaluation	X	X	
Observation of supplier acceptance tests as appropriate	X		
12. Control of Measuring and Test Equipment			
Tools, gauges, and instruments in formal calibration program	X	X	
13. Handling, Storage, and Shipping Control			
Written plans and procedures	X	X	
Routine handling			X
14. Inspection, Test, and Operating Status			
Individual items identified as to status or condition	X	X	
Status indicated by stamps, tags, labels, etc.	X	X	
Visual examination only			X
15. Nonconforming Materials, Parts, or Components			
Written procedures to prevent inadvertent use	X	X	
Nonconformance documented and closed	X	X	
Disposal without records			X
16. Corrective Action			
Conditions adverse to quality identified and corrected	X	X	X
Cause and corrective action documented	X	X	
Safety significant events reported	X	X	X

Table 9.1 Typical Level of QA Effort by Quality Category (cont.)

QA Element/Level of Effort	Category A	Category B	Category C
17. QA Records			
Design and use records	X	X	
Results of reviews, inspections, tests, audits, surveillances, and materials analysis	X	X	
Personnel qualifications	X	X	
Records of fabrication retained for life of package plus 3 years	X	X	
Records of acceptance testing retained for life of package plus 3 years	X	X	
Records of maintenance retained for life of package plus 3 years	X	X	
Shipping records retained for 3 years after shipment	X	X	X
Records managed by a written procedure for retention and disposal	X	X	X
18. Audits			
Written plan of periodic audits	X	X	X
Implementation by written procedures	X	X	X
Lead auditor certified	X	X	
All auditors certified	X		

In discussing the 18 quality criteria and the general areas illustrated in Table 9-1, the SARP should also identify specific QA requirements applicable to:

- Material specifications
- Fabrication specifications
- Operating procedures
- Acceptance tests
- Maintenance program
- Package records.

Requirements for many fabrication processes (e.g., welding, heat treating, and nondestructive examination) are often included in the code or standard used for design and fabrication (and specified on the drawing), and special processes (e.g., pouring lead and resin shielding, applying special coatings, and injecting foam) are generally specified by more detailed procedures to

ensure that the process is appropriately controlled. Similarly, many material requirements may be specified by codes or standards, but some components (e.g., neutron poisons, honeycomb, or special foams) may need to be specified by other means.

QA requirements for all operating procedures and acceptance tests/maintenance program presented in the SARP should be addressed as appropriate. Because the procedures and tests specified in the Operating Procedures chapter and Acceptance Tests and Maintenance Program chapter are those important to the safe operation and performance of the package throughout its service life, each activity described in these chapters of the SARP should generally be subject to the quality assurance requirements of Subpart H, including (but not limited to) written procedures, training of personnel, verification, documentation, nonconformance control, record retention, and audit. Justification should be provided for any activity presented in these chapters that is not subject to Subpart H QA requirements.

Verify that the SARP identifies package records that affect quality. General requirements for package records are specified in §71.91(c), §71.135, and §71.91(a). Additional guidance on types of records that should be retained for each quality category is provided in Chapter 4 of NUREG/CR-6407. Retention periods for records should be consistent with the requirements of §71.91.

The review should also address reporting requirements of §71.95. The QA program should ensure that any occurrence of these events are reported to the DOE Headquarters Certifying Official.

9.3.3 Appendix

Confirm that the appendix includes a list of references, copies of appropriate references not generally available to the reviewer, audit results, and other appropriate supplemental information. Detailed QA procedures should not be provided in the SARP but may be requested during the SARP review.

9.4 Evaluation Findings

9.4.1 Findings

The reviewer should ensure that the information presented supports a conclusion that the regulatory requirements in Section 9.3.2 above are satisfied.

The TRR should include a finding similar to the following:

> Based on review of the statements and representations in the SARP, the staff concludes that the quality assurance program has been adequately described and meets the quality assurance requirements of 10 CFR Part 71. Package-specific requirements are adequate to assure that the package is designed, fabricated, assembled, tested, used, maintained, modified, and repaired in a manner consistent with its evaluation.

9.4.2 Conditions of Approval

The TRR should clearly identify any conditions of approval that should be included in Section 5 of the certificate of compliance. In addition to information specified on the package drawings, operating procedures, and acceptance tests/maintenance program, other conditions of approval that may be applicable to the Quality Assurance chapter of the SARP include those items discussed in Section 9.2 above.

Care should be taken to ensure that conditions of approval apply to all organizations that may be involved in packaging activities. Conditions of approval should not include site-specific requirements or procedures.

9.5 References

1. Department of Energy, "Quality Assurance," DOE O 414.1, November 24, 1998.
2. American Society of Mechanical Engineers, "Quality Assurance Requirements for Nuclear Facility Applications," ASME NQA-1-1997 Edition, December 31, 1997, New York, New York.
3. U.S. Nuclear Regulatory Commission, "Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material," Regulatory Guide 7.10, Rev. 1.
4. U.S. Nuclear Regulatory Commission, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," NUREG/CR-6407 (INEL-95/0551), February 1996.

APPENDIX A:
DEFINITIONS

The majority of package terms are defined in 10 CFR 71.4 or 49 CFR 173.403, and are repeated in Table A.1 for convenience. Where applicable, the source of each definition is indicated. In many cases, terms defined in 10 CFR 71.4 are also defined in 49 CFR 173.403.

Table A.1 Definitions

A_1	The maximum activity of special form radioactive material permitted in a Type A package. [10 CFR 71.4]
A_2	The maximum activity of radioactive material, other than special form, low specific activity, and surface contaminated object material, permitted in a Type A package. [10 CFR 71.4]
Carrier	A person engaged in the transportation of passengers or property by land or water as a common, contract, or private carrier, or by civil aircraft. [10 CFR 71.4]
Certificate holder	A person who has been issued a certificate of compliance or other package approval. [10 CFR 71.4]
Certificate of compliance	A certificate issued by DOE approving for use, with specified limitations, a specific packaging. Certificates of compliance are also issued by NRC.
Close reflection by water	Immediate contact by water of sufficient thickness for maximum reflection of neutrons. [10 CFR 71.4]
Closed transport vehicle	A transport vehicle or conveyance equipped with a securely attached exterior enclosure that during normal transportation restricts the access of unauthorized persons to the cargo space containing the Class 7 (radioactive) materials. The enclosure may be either temporary or permanent, and in the case of packaged materials may be of the "see-through" type, and must limit access from the top, sides, and bottom. [49 CFR 173.403]
Containment system	The assembly of components of the packaging intended to retain the radioactive material during transport. [10 CFR 71.4]
Conveyance	For transport by public highway or rail, any transport vehicle or large freight container; for transport by water, any vessel or any hold, compartment, or defined deck area of a vessel, including any transport vehicle on board the vessel; and for transport by aircraft, any aircraft. [10 CFR 71.4]
Damaged fuel	Fuel with known or suspected cladding defects greater than a hairline crack or a pinhole leak.

Exclusive use	The sole use by a single consignor of a conveyance for which all initial, intermediate, and final loading and unloading are carried out in accordance with the direction of the consignor or consignee. The consignor and the carrier must ensure that any loading or unloading is performed by personnel having radiological training and resources appropriate for safe handling of the consignment. The consignor must issue specific instructions, in writing, for maintenance of exclusive use shipment controls, and include them with the shipping paper information provided to the carrier by the consignor. [10 CFR 71.4]
Fissile material	Plutonium-238, plutonium-239, plutonium-241, uranium-233, uranium-235, or any combination of these radionuclides. Unirradiated natural uranium and depleted uranium, and natural uranium or depleted uranium that has been irradiated in thermal reactors only are not included in this definition. Certain exclusions from fissile material controls are provided in 10 CFR 71.53. [10 CFR 71.4]
Fissile material package	A fissile material packaging together with its fissile material contents. [10 CFR 71.4]
Low specific activity (LSA) material	Radioactive material with limited specific activity that satisfies the descriptions and limits specified in 10 CFR 71.4.
Maximum normal operating pressure	The maximum gauge pressure that would develop in the containment system in a period of one year under the heat condition specified in 10 CFR 71.71(c)(1), in the absence of venting, external cooling by an ancillary system, or operational controls during transport. [10 CFR 71.4]
Natural thorium	Thorium with the naturally occurring distribution of thorium isotopes (essentially 100 weight percent thorium 232). [10 CFR 71.4]
Normal form radioactive material	Radioactive material that has not been demonstrated to qualify as "special form radioactive material." [10 CFR 71.4]
Optimum interspersed hydrogenous moderation	The presence of hydrogenous material between packages to such an extent that the maximum nuclear reactivity results. [10 CFR 71.4]
Package	The packaging together with its radioactive contents as presented for transport. [10 CFR 71.4]

Packaging	The assembly of components necessary to ensure compliance with the packaging requirements of 10 CFR Part 71. It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding, and devices for cooling or absorbing mechanical shocks. The vehicle, tie-down system, and auxiliary equipment may be designated as part of the packaging. [10 CFR 71.4]
Quality assurance	All planned and systematic actions necessary to provide adequate confidence that a system or component will perform satisfactorily in service. [10 CFR 71.101]
Radiation level	The radiation dose-equivalent rate expressed in millisievert(s) per hour or mSv/h (millirem(s) per hour or mrem/h). Neutron flux densities may be converted into radiation levels according to Table 1, 49 CFR 173.403. [49 CFR 173.403]
Radioactive contents	The radioactive material within the package containment system. [49 CFR 173.403]
Radioactive material	Any material having a specific activity greater than 70 Bq per gram (0.002 microcurie per gram). [49 CFR 173.403]
Reference air leakage rate	The allowable leakage rate converted to reference cubic centimeters per second. [ANSI N14.5]
Reference cubic centimeter per second (ref cc/s)	A volume of one cubic centimeter of dry air per second at one atmosphere absolute pressure (760 mm Hg) and 25°C. [ANSI N14.5]
Safety Evaluation Report (SER)	A report issued by the DOE Headquarters Certifying Official that documents DOE's review of the package for compliance with DOE O 460.1A and 10 CFR Part 71.
Special form radioactive material	Radioactive material that satisfies the conditions specified in 10 CFR 71.4.
Specific activity of a radionuclide	The radioactivity of the radionuclide per unit mass of that nuclide. The specific activity of a material in which the radionuclide is essentially uniformly distributed is the radioactivity per unit mass of the material. [10 CFR 71.4]
Surface contaminated object (SCO)	A solid object that is not itself classed as radioactive material, but which has radioactive material distributed on any of its surfaces. SCO must be in one of two groups with surface activity not exceeding the limits specified in 10 CFR 71.4.

Technical Review Report (TRR)	A report prepared by the DOE review staff that documents the technical review of the package for compliance with DOE O 460.1 and 10 CFR Part 71. The TRR provides the justification for the technical information included in the SER.
Transport index (TI)	The dimensionless number (rounded up to the next tenth) placed on the label of a package, to designate the degree of control to be exercised by the carrier during transportation. The transport index is determined as follows: (1) for non-fissile material packages, the number determined by multiplying the maximum radiation level in millisievert (mSv) per hour at one meter (3.3 ft) from the external surface of the package by 100 (equivalent to the maximum radiation level in millirem per hour at one meter (3.3 ft)); or (2) for fissile material packages, the number determined by multiplying the maximum radiation level in millisievert per hour at one meter (3.3 ft) from the external surface of the package by 100 (equivalent to the maximum radiation level in millirem per hour at one meter (3.3 ft)), or, for criticality control purposes, the number obtained as described in 10 CFR 71.59, whichever is larger. [10 CFR 71.4]
Type A quantity	A quantity of radioactive material, the aggregate radioactivity of which does not exceed A_1 for special form radioactive material, or A_2 for normal form radioactive material, where A_1 and A_2 are given in Table A-1 of 10 CFR Part 71, or may be determined by procedures described in Appendix A of 10 CFR Part 71. [10 CFR 71.4]
Type A packaging	A packaging approved to transport a Type A quantity of radioactive contents.
Type B package	A Type B packaging together with its radioactive contents. On approval, a Type B package design is designated as B(U) unless the package has a maximum normal operating pressure of more than 700 kPa (100 psi) gauge or a pressure relief device that would allow the release of radioactive material to the environment under the tests specified in §71.73 (hypothetical accident conditions), in which case it will receive a designation B(M). B(U) refers to the need for multilateral approval of international shipments. There is no distinction made in how packages with these designations may be used in domestic transportation. To determine their distinction for international transportation, see DOT regulations in 49 CFR Part 173. A Type B package approved before September 6, 1983 was designated only as Type B. [10 CFR 71.4]
Type B packaging	A packaging approved to transport a Type B quantity of radioactive contents.

Type B quantity	A quantity of radioactive material greater than a Type A quantity. [10 CFR 71.4]
Uranium–natural	Uranium with the naturally occurring distribution of uranium isotopes (approximately 0.711 weight percent uranium-235, and the remainder essentially uranium-238). [10 CFR 71.4]
Uranium–depleted	Uranium containing less uranium-235 than the naturally occurring distribution of uranium isotopes. [10 CFR 71.4]
Uranium–enriched	Uranium containing more uranium-235 than the naturally occurring distribution of uranium isotopes. [10 CFR 71.4]

APPENDIX B:
SUMMARY LISTING OF 10 CFR PART 71 REQUIREMENTS

This appendix provides a summary listing of the sections in 10 CFR Part 71 and the primary sections of the PRG to which they apply. In several cases, the applicability is a subjective judgment, which may depend on the package design as well as on the specific format in which the SARP is organized. The user is cautioned accordingly.

Table B.1 Summary Listing of 10 CFR Part 71 Requirements

Section/ Chapter	General Information	Structural	Thermal	Containment	Shielding	Criticality	Op. Proc.	Acc. Tests & Maint.	Quality Assurance	Comments
71.0(d)	X									Application for package approval
71.13	X						X			Previously approved package
71.31(a)(1)	X	X	X	X	X	X				Package description
71.31(a)(2)		X	X	X	X	X				Package evaluation
71.31(a)(3)	X								X	Description of QA program
71.31(b)	X									See also 71.13 for grandfathering
71.31(c)	X	X	X	X	X	X	X	X	X	Identification of codes and standards. Primary interest is ASME B&PV Code but applicable to ANSI N14.5 and perhaps others
71.33	X	X	X	X	X	X				Packaging and content description
71.35(a)		X	X	X	X	X				Package evaluation
71.35(b)	X					X				Max. packages/shipment based on criticality
71.35(c)							X			Special fissile material controls
71.37									X	QA
71.38	X									Renewal of certificate or QA program approval
71.39										Any additional information may be required.

Table B.1 Summary Listing of 10 CFR Part 71 Requirements (cont.)

Section/ Chapter	General Information	Structural	Thermal	Containment	Shielding	Criticality	Op. Proc.	Acc. Tests & Maint.	Quality Assurance	Comments
71.41(a)		X	X							Demonstration of compliance
71.41(b)										Vehicle may be considered in evaluation.
71.41(c)										Variations in §§71.71 and 71.73 may be approved by NRC.
71.43(a)	X									Size
71.43(b)	X									Tamper-indicating device
71.43(c)				X						Positive closure
71.43(d)		X	X	X						Chemical or galvanic reactions
71.43(e)				X						Valves
71.43(f)		X	X	X	X	X				Package effectiveness
71.43(g)	X		X							Temperature limits
71.43(h)				X						Venting
71.45		X								Lifting and tie-down
71.47(a)	X				X		X			Dose rates, nonexclusive use
71.47(b)	X				X		X			Dose rates, exclusive use
71.47(c)							X			Instructions for exclusive use shipments
71.47(d)							X			Instructions for exclusive use shipments

Table B.1 Summary Listing of 10 CFR Part 71 Requirements (cont.)

Section/ Chapter	General Information	Structural	Thermal	Containment	Shielding	Criticality	Op. Proc.	Acc. Tests & Maint.	Quality Assurance	Comments
71.51(a)(1)		X	X	X	X	X				NCT leakage, shielding, package effectiveness
71.51(a)(2)				X	X					HAC leakage and shielding
71.51(b)	X			X						A_2 for mixture
71.51(c)			X	X						Filters and mechanical cooling
71.52										Exemptions for LSA
71.53										Fissile exempt quantities
71.55(a)										Criticality, general
71.55(b)						X				Water inleakage analysis
71.55(c)										Exemption from water inleakage
71.55(d)		X				X				NCT criticality
71.55(e)						X				HAC criticality
71.57										Reserved
71.59	X					X				Criticality, arrays
71.61		X								Deep water immersion for spent fuel only
71.63	X			X						Special containment for Pu
71.64										Pu air shipment
71.65										Any other requirements may be imposed to protect public health or minimize danger to life or property.

Table B.1 Summary Listing of 10 CFR Part 71 Requirements (cont.)

Section/ Chapter	General Information	Structural	Thermal	Containment	Shielding	Criticality	Op. Proc.	Acc. Tests & Maint.	Quality Assurance	Comments
71.71		X	X							NCT tests
71.73		X	X							HAC tests
71.74										HAC tests for Pu air shipments
71.75										Special form
71.77										LSA-III
71.81							X		X	Operating controls
71.83						X	X			Assumptions for unknown properties
71.85(a)								X		Cracks, voids
71.85(b)		X						X		Pressure test
71.85(c)	X						X	X		Data plate
71.87(a)							X			Proper contents
71.87(b)							X	X		Undamaged packaging
71.87(c)							X			Closure devices
71.87(d)							X			Liquid systems
71.87(e)							X			Pressure relief devices
71.87(f)							X			Loaded by procedures
71.87(g)							X	X		Moderator/absorber present
71.87(h)							X			Tie-down devices

Table B.1 Summary Listing of 10 CFR Part 71 Requirements (cont.)

Section/ Chapter	General Information	Structural	Thermal	Containment	Shielding	Criticality	Op. Proc.	Acc. Tests & Maint.	Quality Assurance	Comments
71.87(i)							X			Non-fixed contamination
71.87(j)							X			Radiation levels
71.87(k)							X			Surface temperatures
71.88										Pu air shipment
71.89							X			Opening instructions
71.91									X	Records
71.93								X		Inspection and tests
71.95									X	Reports of problems
71.97										Advance notification of spent fuel and HLW shipments
71.99										Violations
71.101-137	X								X	QA (Subpart H)

APPENDIX C:
SUMMARY OF CHANGES RESULTING FROM THE 1996 (AS AMENDED) REVISION OF 10 CFR PART 71

The attached table summarizes changes resulting from the 1996 revision of 10 CFR Part 71. The primary purpose of this revised rule was to conform NRC regulations with those of the International Atomic Energy Agency.*

Package designs that satisfy the 1996 revision of 10 CFR Part 71 are designated with the identification number suffix "-85." The changes listed in this appendix are applicable to all packages with initial approval after April 1, 1996, and to other applications requesting the addition of the "-85" suffix. Because DOE generally expects that its packages comply with the most current regulations, these changes should also be addressed during the re-certification of previously approved DOE packages.

Subsequent to the 1996 revision of 10 CFR Part 71, two changes have been promulgated: (1) several additional restrictions for fissile material exemptions and general license provisions, and (2) an additional exemption from the double containment requirements for plutonium. These changes are also addressed in the table below.

Changes in the following general areas are *excluded* from the table because they are seldom applicable to packages certified by DOE: limited specific activity (LSA), surface contaminated objects (SCO), air shipments of plutonium, and special form qualification. The reviewer is cautioned that if these areas are applicable to the package, the changes may be very significant.

Based on review experience to date, the following changes to 10 CFR Part 71 appear to be the most significant for packages reviewed by DOE:

- Reflection requirements for the criticality analysis of the containment system of a single package, §71.55(b)(3)
- Replacement of Fissile Class by a transport index based on criticality control, and a possible change in the number of packages that must be analyzed in an array of previous Fissile Class III or Fissile Class I packages, §71.59 and §71.4
- Requirement for dynamic crush test of certain lightweight, low-density packages with significant quantities of radioactive material, §71.73(c)(2)
- Thermal test requirements under hypothetical accident conditions, §71.73(c)(4)
- Reduction in A_2 value for uranium enriched between 5% and 20%, Table A-1.

* Safety Series No. 6, *Regulations for the Safe Transport of Radioactive Material 1985 Edition (As Amended 1990)*, International Atomic Energy Agency, Vienna, 1990.

Table C.1 Summary of Changes Resulting from the 1996 (as Amended) Revision of 10 CFR Part 71

Part 71 1996	Part 71 1983	Change	Comments
71.4	71.53	Natural or depleted uranium irradiated in a thermal reactor only is now defined *not* to be fissile material. Previously it was considered fissile exempt.	No significant impact is apparent.
71.13	71.13	Provisions for previously approved packages have been updated to reflect revised rule implementation. Additional restrictions based on criticality now apply for addition of new contents to a previously approved package. Packages now need serial numbers.	Fabrication of packages without "-85" designation must have been completed by 4/1/99.
71.18	71.18	Additional restrictions on special moderators have been included in general license provisions.	Emergency rule change effective Feb. 97
71.22	71.22	Additional restrictions on special moderators have been included in general license provisions.	Emergency rule change effective Feb. 97
71.43(c)	71.43(c)	Containment standards have been modified to add a requirement that the positive fastening device cannot be opened by pressure that may arise within the package.	No significant impact is apparent.
71.43(g)	71.43(g)	Allowed package surface temperature for exclusive-use shipments has been increased to 85°C (185°F) from 82°C (180°F).	No significant impact is apparent.
71.47(b)	71.47(a)	Allowable radiation levels for exclusive-use shipments are now defined for external surface of package rather than for accessible external surface.	No significant impact is apparent. Personnel barriers can still be considered part of the package if analyzed appropriately. Required analysis may increase.
71.51(a)(2)	71.51(a)(2)	Allowable release of Kr-85 under hypothetical accident conditions has been decreased from 10,000 Ci to 2700 Ci (10 A_2)	Impact is not apparent. Kr-85 is significant nuclide generally only for spent fuel or fission product contents.

Table C.1 Summary of Changes Resulting from the 1996 (as Amended) Revision of 10 CFR Part 71 (cont.)

Part 71 1996	Part 71 1983	Change	Comments
71.53	71.53	Additional restrictions on fissile material exemptions have been included.	Emergency rule change effective Feb. 97.
71.55(b)(3)	71.55(b)(3)	Full reflection must now be applied to the containment system itself as well as to the package (whichever is more reactive).	Possibly significant for criticality analysis of some single packages.
71.59	71.57 71.59 71.61	Fissile Class designation has been replaced by a criticality transport index similar to that for previous Fissile Class II. Transport index now controls the number of packages allowed in one shipment and the method of transport (e.g., exclusive use) required.	Major impact is for Fissile Class III. Demonstration of minimum requirements under 1983 rule will not satisfy 1996 requirements. Minor impact is also possible for previous Fissile Class I.
71.61	N/A	Deep-water immersion design requirement has been added for spent fuel packagings with activity greater than 37 PBq.	Adopted with wording changes from IAEA requirement. Most significant impact is likely to result from restriction on inleakage of water.
71.63	71.63	Vitrified glass has been exempted from double containment under certain conditions.	Final rule change effective July 98.
71.71(c)(7)	71.71(c)(7)	Requirement that free drop of Fissile Class II packages be preceded by corner drop has been removed.	Fissile Class II is no longer defined. Additional corner drop requirements for fissile material packages have been added to 71.71(c)(8).
71.71(c)(8)	71.71(c)(8)	Requirement for corner drop of lightweight packages has been expanded to include fissile material packages.	Overall effect of changes to 71.71(c)(7) and 71.71(c)(8) appears small.
71.73(c)(2)	N/A	Requirement for dynamic crush test has been added for certain lightweight, low-density packages with large quantities of radioactive material.	If applicable, change appears very significant.

Table C.1 Summary of Changes Resulting from the 1996 (as Amended) Revision of 10 CFR Part 71 (cont.)

Part 71 1996	Part 71 1983	Change	Comments
71.73(c)(4)	71.73(c)(3)	Requirements for fuel source geometry have been added. Actual convection during fire and insolation after fire must also be considered.	Generally requires new analysis of previously approved packages. Insolation discussed in Federal Register Notice of rule change (FR, 9/28/95), but not in 71.73 itself.
71.73(c)(5)	71.73(c)(4)	Time period for the 0.9-m immersion test of fissile packages has been eliminated.	No significant impact is apparent, but length of test may not be considered to limit amount of water inleakage.
71.73(c)(6)	71.73(c)(5)	Time period for the 15-m immersion test of all packages has been eliminated.	No significant impact is apparent, but length of test may not be considered to limit water inleakage.
71.85(c)	71.85(c)	Requirement for serial numbers on packages has been added.	See 71.13. The requirement for serial numbers also applies to previously approved packages.
Appendix A (II), and Table A-2	Appendix A (I)(2), I(3), II(1)	Determination of A_1 or A_2 values for nuclides not listed in Table A-1 must now be approved by NRC unless general values from Table A-2 are used. Previous rule provided method to calculate A_1 and A_2 that did not require NRC approval, as well as other general values.	No significant impact is apparent. Calculation of A_1 or A_2 values for such nuclides was rarely done by applicant. If applicable, needs to be checked carefully.
Table A-1	Table A-1	Numerous A_1 and A_2 values have been changed to adopt IAEA values (in SI units). Total number of nuclides listed increased from 284 to 378. 144 A_1 values increased, 73 decreased. 129 A_2 values increased, 95 decreased.	All A_1 and A_2 values for Pu increased. Almost all A_1 values for U increased, most A_2 values decreased. Enriched U with unlimited A_1 and A_2 values now restricted to ≤5%, rather than <20%. Latter change appears significant.

APPENDIX D:
MATERIALS AND FABRICATION

Issues related to package materials and fabrication are interlaced among all chapters in the SARP. Although some aspects of the review are relatively straightforward (e.g., thermal properties of materials should be discussed in the Thermal Evaluation chapter), other issues may not be clearly aligned with the nine chapters of the SARP format. Consequently, the review of material and fabrication should address all SARP chapters to ensure that these areas have been properly evaluated.

Tables D.1 and D.2 provide a summary of typical issues that should be reviewed for materials and fabrication, respectively. The reviewer is cautioned not to use these tables as a simple "yes or no" checklist, but to consider each package and its specific issues on a case-by-case basis.

As noted in Chapter 1 of this PRG, information on materials and fabrication which is indicated on engineering drawings may be described in additional detail in a separate fabrication specification.

Table D.1 Review of Materials

Identification of Packaging Components

- Is each packaging component depicted on the drawings and identified in the parts list or by other appropriate means?
- Is each packaging component not identified on the drawings properly justified as not important to safety?

Material Specifications of Packaging Components

- Is the material of construction of each packaging component specified on the drawings?
- Is a material specification (e.g., ASME, ASTM, commercial equivalent) designated on the drawings for each material? Is the material specification appropriate for the code or standard applicable to the packaging?
- For materials without an applicable specification, are material properties to be controlled properly specified on the drawings? Examples include minimum/maximum densities of foam, fiberboard, and similar materials, and minimum density neutron absorbing nuclides. Are these properties consistent with those used in the package evaluation?
- Are appropriate examination requirements for each material specified on the drawings?

Material Properties

- Are material properties relevant to the SARP evaluation specified where appropriate?
- Are the material properties appropriate for the temperatures and pressures under normal conditions of transport and hypothetical accident conditions?
- Have appropriate test requirements for materials been established?

Brittle Fracture

- Is any packaging material subject to brittle fracture by cold or other mechanisms (e.g., hydrogen embrittlement)?
- Are the criteria of RG 7.11 or 7.12 satisfied?
- Has embrittlement by other mechanisms (e.g., fabrication processes) been properly addressed?

Chemical, Galvanic, and Other Reactions

- Is any material subject to chemical, galvanic, or other reaction (e.g., radiolysis) with each other or with the contents? If so, have these issues been properly addressed in the package evaluation?
- Is any material subject to radiation damage? If so, has this issue been properly addressed?

Operating Procedures

- Should any material or component be inspected and/or replaced prior to each use?
- Are appropriate types of inspections and acceptance criteria specified?

Acceptance Testing and Maintenance Program

- Should any material or component be subject to acceptance testing prior to first use?
- Should any material or component be inspected, maintained, and/or replaced as part of a periodic maintenance program? Is the period and type of inspection appropriate? Is the maintenance or replacement schedule appropriate?
- Are the requirements for acceptance testing and maintenance specified?

Quality Assurance

- Has each component been properly categorized as to its importance to safety?
- Have appropriate controls been established in the Quality Assurance chapter to assure that quality requirements are met?
- Has appropriate documentation been specified to document that quality requirements are met?

Table D.2 Review of Fabrication

Identification of Packaging Components

- Is each packaging component depicted on the drawings and identified in the parts list or by other appropriate means?
- Is each packaging component not identified on the drawings properly justified as not important to safety?

Welds

- Is the location, type, size, filler material, post-weld treatment process (if applicable), and method of examination for each weld specified on the drawings by an appropriate welding symbol?
- Is a code or standard for each weld and welding procedure specified on the drawings? Is the weld information consistent with this code or standard?
- Is the code or standard for the weld appropriate?

Codes and Standards for Other Fabrication Processes

- Is an appropriate code or standard for fabrication of each packaging component specified on the drawings?
- For components without an applicable specification (e.g., lead shielding), is the fabrication process sufficiently described, controlled, and specified on the drawings?
- Are appropriate examination requirements for each fabrication process specified on the drawings?
- Is the package evaluation consistent with its fabrication specifications?

Operating Procedures

- Should components or features be inspected prior to each use?
- Are appropriate types of inspections and acceptance criteria specified?

Acceptance Testing and Maintenance Program

- Are appropriate acceptance tests specified to address fabrication issues (e.g., uniformity of lead)?
- Should any component or feature be inspected, maintained, and/or replaced as part of a periodic maintenance program? Is the period and type of inspection appropriate? Is the maintenance or replacement schedule appropriate?
- Are the requirements for acceptance testing and maintenance specified?

Quality Assurance

- Has each component been properly categorized as to its importance to safety?
- Are training requirements for fabrication personnel properly specified?
- Have appropriate controls been established in the Quality Assurance chapter to assure that quality requirements are met?
- Has appropriate documentation been specified to document that quality requirements are met?

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